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NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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US-APWR SUBCOMMITTEE

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WEDNESDAY

NOVEMBER 20, 2013

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ROCKVILLE, MARYLAND

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

COMMITTEE MEMBERS:

JOHN W. STETKAR, Subcommittee Chairman

J. SAM ARMIJO, Member

CHARLES H. BROWN, JR. Member

HAROLD B. RAY, Member

PETER C. RICCARDELLA, Member

STEPHEN P. SCHULTZ, Member

DESIGNATED FEDERAL OFFICIAL:

GIRIJA S. SHUKLA

ALSO PRESENT:

EDWIN M. HACKETT, Executive Director, ACRS
SARDAR AHMED, NRO
MARK BIERY, MNES
PERRY BUCKBERG, NRO
THERESA CLARK, NRO
JOHN CONLY, Luminant
JENNIFER DIXON-HERRITY, NRO
TODD EVANS, Luminant
DENNIS GALVIN, NRO
RAUL HERNANDEZ, NRO
THOMAS HICKS, MNES
JOHN HONCHARIK, NRO
MASASHI ITO, MNES
PAUL KALLAN, NRO
PETER KANG, NRR
RON LAVERA, NRO
RENEE LI, NRO
ATSUSHI MATSUMOTO, MHI
YOSHIHIRO MINAMI, MHI
STEPHEN MONARQUE, NRO
MASATOSHI NAGAI, MNES

HIROKI NISHIO, MHI

YUKIHIRO NISHIZAWA, MHI

RYAN NOLAN, NRO

HIKARU OGASAWARA, MHI

RICHARD SAMPLES, MNES

THOMAS SCARBROUGH, NRO

RYAN SPRENGEL, MNES

ANGELO STUBBS, NRO

YOSHIHIRO TAKAYAMA, NRO

LARRY WHEELER, NRO

DONALD WOODLAN, Luminant

*Present via telephone

T-A-B-L-E O-F C-O-N-T-E-N-T-S

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

(8:33 a.m.)

CHAIRMAN STETKAR: The meeting will now come to order. This is the meeting of the United States Advanced Pressurized Water Reactor Subcommittee, I'm John Stetkar, Chairman of the Subcommittee Meeting.

ACRS members in attendance are Pete Riccardella, Harold Ray, Steve Schultz, Sam Armijo, and Charles Brown. I'm not sure whether we'll be joined by Ron Ballinger or not.

Mr. Girija Shukla of the ACRS staff is the designated Federal Official for this meeting. The Subcommittee will discuss Chapter 3, Design of Structure Systems, Components, and Equipment, except for Sections 3.7 and 3.8 of the Safety Evaluation Report associated with the US-APWR design certification and the Comanche Peak Combined License Application.

The Subcommittee will also discuss Chapter 9, Auxiliary Systems for the Comanche Peak Combined License Application. We will hear presentations from Mitsubishi Heavy Industries, Mitsubishi Nuclear Energy Systems and the NRC staff, and Luminant.

The Subcommittee will gather information, 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 Mitsubishi 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 discussions and to verify that only people with the required clearance and need-to-know are present.

So if we do tread into proprietary information, please alert me and we'll make sure that we close the meeting. I'll rely on you to do that. A transcript of the meeting is being kept and will be made available as stated in the Federal Register Notice.

Therefore, we request the participants in this meeting use the microphones located throughout the meeting room when addressing the Subcommittee. The participant should first identify themselves and speak with sufficient clarity and volume so they may be readily heard.

A telephone bridge line has also been established for the meeting. To preclude interruption

of the meeting the phone will be placed in a listen-in mode during the presentations and committee discussions. We'll open it up if you need any assistance form anyone.

Please silence your cell phones and we'll now proceed with the meeting and I'll ask NRC Management, Jennifer?

MS. DIXON-HERRITY: Yes.

CHAIRMAN STETKAR: Thank you.

MS. DIXON-HERRITY: Good morning. My name is Jennifer Dixon-Herrity. I'm the Chief of the Licensing Branch that's overseeing the NRC review for the US-APWR Design.

To summarize the US-APWR review to this point, staff has issued Safety Evaluation Reports with open items addressing 16 chapters of the Application. Of those issued chapters all but today's partial, Chapter 3, Chapter 6, and Chapter 7, have been presented to the ACRS Full Committee. Chapters 6 and 7 are going to be presented on December 5 of this year.

Today's meeting will cover the partial Chapter 3 with SE with open items. The partial Chapter 3 was issued because the evaluations for Section 3.7 and 3.8 are currently on critical path for this review and were separated to allow this Chapter to move forward.

And with that, we can probably move on.

CHAIRMAN STETKAR: Great. Thank you very much and I guess I'll turn it over to Ryan Sprengel of MNES.

MR. SPRENGEL: Thank you, good morning everyone. Once again, I think we're all glad to be back for another meeting. As most or many of you know, MHI has recently communicated to the staff that there will be a slowdown of the US-APWR Design Certification effort.

MHI does remain committed to the US-APWR Design Certification and we will maintain activities going forward, albeit at a slightly different pace. We do continue to make significant progress and I think we've had many, many meetings, and many positive meetings, and we do want to maintain this effort and continue to interact with both the staff for the reviews and the ACR Subcommittee and Full Committee.

Right now we're still in the discussion and planning mode, both internally and with the staff, so I'd request that we delay any detail discussion on any potential impacts.

We'll maintain communication with the ACRS going forward and communicate where we stand in the future.

CHAIRMAN STETKAR: Yes, and we, as long as, you know, we get far enough warning we can accommodate

your schedule in our Subcommittee meetings and Full Committee meetings, so we'll just adjust as necessary as we go forward.

MR. SPRENGEL: Thank you. So there may be a few times in the meeting when some activities or items come up for discussion and we'll identify them clearly and let you know what the situation is, but we may not have all the answers right now.

CHAIRMAN STETKAR: Sure.

MR. SPRENGEL: Okay. I'll go ahead and turn it over to Tom Hicks.

MR. HICKS: Great, thanks, Ryan. Good morning, my name is Tom Hicks. I'm with MNES Licensing. I'm joined by Nagai-san from MNES and Ogasawara-san and Nishio-san from MHI.

As was discussed, we'll be presenting US-APWR DCD Chapter 3 today with the exceptions of 3.7 and 3.8. I'll be discussing Sections 3.1, 3.2, and 3.11, and then we'll do some seat switching and Mark Biery will step up and present the remaining Chapter 3 sections that we're going to talk about today.

We will change some of the MHI support staff as we proceed through the presentation. We'll try not to be disruptive when we do that, but thanks a lot.

These are the, the list of the people that

we'll be calling upon possibly during the meeting and the various sections in Chapter 3 that we'll be discussing today.

And 3.1 is the first section that we'll talk about which is NRC General Design Criteria, Performance with those Criteria. Section 3.1 of the DCD provides a high-level description of how the general design criteria are met and what specific DCD sections provide further technical discussion regarding each of those criteria.

In general, conformance with the applicable GDC criteria discussed in each individual section. There are no open items in Section 3.1. Next Section is 3.2 which is Classification of Structured Systems and Components.

The US-APWR DCD Section 3.2 describes classification of SSCs according to nuclear safety classification, seismic category, quality groups, quality assurance classification, and codes and standards.

Subsection 3.2.1 describes the classification of SSCs in terms of seismic category. Subsection 3.2.2 describes the various equipment classifications which segregates systems and components based on safety classification, seismic category,

quality groups, quality assurance classification, and applicable codes and standards.

And there is one open item associated with Section 3.2 that we'll discuss at the end. The DCD uses the guidance provided in the NRC Reg Guide 1.29 to meet GDC 2 for identifying and classifying those SSCs and will have design to withstand the effects of earthquakes without loss of capability to perform safety functions. These SSCs are classified as Seismic Category I.

SSCs that are not required to remain functional following an SSC, but whose failure could degrade performance of safety-related SSCs to an unacceptable safety level are classified as Seismic Category II.

And Seismic Category II SSCs are designed and constructed to maintain their structural integrity under seismic loading from an SSC. A description of the specific analysis test performed for Seismic I and II SSCs is in 3.7 which we're not going to get into today.

Other systems that fall outside the standard Seismic Category I and II definitions but receive special seismic design requirements are also specified in the DCD.

Some of these special seismic requirements are driven by NRC Regulatory Guides, and these systems

include the safety-related instrument lines which refers to, which gets requirements from Reg Guide 1.151, fire protection systems, which has requirements in Reg Guide 1.189, rad waste management systems, Reg Guide 1.143, and then the diverse automatic actuation system.

SSCs that are not classified either Seismic Category I and II are classified as non-seismic, or NS. Quality groups, under 50.55(a), Codes and Standards, certain systems and components of nuclear power reactors must be designed, fabricated, erected, and tested in accordance with the standards for Class I, II, and III components given in Section III of the ASME Code, or an equivalent standard.

Class I components are designated as Quality Group A. Reg Guide 1.26 describes the quality classification system related to specific National Standards that may be used to determine quality standards applicable to the NRC for satisfying GDC 1.

And for other safety-related components containing water, steam, radioactive material in light water cold nuclear power plants. Class II components are included in Quality Groups B and Class III included in Quality Group C.

Quality Group D includes water and steam containing components that are not part of the reactor

coolant pressure boundary or included in Quality Groups B or C, but are part of systems, or portions of systems, that contain or may contain radioactive material.

Subsection 3.22 of the DCD describes how Reg Guide 1.26 is applied to the US-APWR, and the Quality Group Classifications described in the DCD meet Reg Guide 1.26.

Equipment classification designation is used to identify the quality and code requirements applicable to a specific component. Systems and components are assigned to an equipment class based on safety function, quality group designation, seismic requirements, quality assurance requirements, and impact on plan availability.

Safety-related equipment is designated as Equipment Classes I through III. Water and steam containing non-safety-related components that are not part of the reactor coolant pressure boundary are not included in Quality Groups B or C or the rad waste management systems, are part of systems or portions of systems that contain or may contain radioactive material and require augmented quality or designated Equipment Class IV.

Some Quality Group D components are included in this category. The codes and standards

identified in Reg Guide 1.26 for Quality Group D are assigned to these components.

Augmented quality assurance requirements for non-safety-related SSCs, as described in the Quality Assurance Program, are applied to Equipment Class IV SSCs. Equipment Class V components include those that are risk significant are designated to meet special seismic requirements such as Seismic Category II, or perform functions that address ATWS or station blackout and are not within the purview of Equipment Classes IV, VI, VII, and VIII.

Augmented quality assurance requirements are applied to Equipment Class V SSCs. Codes and standards apply to these components are defined in the DCD for each system.

Equipment Class VI is assigned to the non-safety-related components of the rad waste management system and parts of the steam generator blowdown system which are outside the containment isolation valves.

Augmented quality assurance requirements are applied to Equipment Class VI SSCs. Codes and standards identified in Reg Guide 1.143 are applied to these components. Equipment Class VII is assigned to non-safety-related components of the fire protection

system.

Augmented quality assurance requirements are applied to Equipment Class VII SSCs and the codes and standards identified in Reg Guide 1.189, Section 1.7 and NFPA-804 are applied to this system.

Equipment Class VIII is assigned to non-safety-related components that contain, or may contain, radioactive materials that are classified as Quality Group D, but not included in Equipment Classes I through VII and do not require augmented quality.

Applicable codes and standards are the same as those applied to Equipment Class IV which come from Reg Guide 1.26. Equipment Classes IX and X include non-safety-related components in structures that do not fall into any one of the Quality Groups A through D or Equipment Classes I through VIII.

And components that have impact on continuous power generation, those are in Class IX and then all the other components are in Class X. DCD Table 3.2-2 provides a summary of the equipment classes, quality assurance classification, applicable codes and standards, and seismic category for US-APWR mechanical and fluid systems, components, and equipment.

As I said earlier, there is one open item in Section 3.2, it has to do with providing information

to the staff regarding when certain design basis documents will be ready for staff audit and MHI is currently preparing a response to the staff regarding this particular open item.

CHAIRMAN STETKAR: Tom, before we leave Section 3.2, this is in some sense a general question, but I'd like to use specific examples to illustrate it.

How do the equipment classifications, you've summarized this morning kind of the basis of the ten different groups, if you will, and different quality attributes and seismic classifications, how do you treat important to safety equipment as identified in Table 17.4-1 of the DCD, which is the Table that identifies, it's called Risk Significant --

MR. HICKS: Risk Significant, right.

CHAIRMAN STETKAR: -- Equipment there, but it's essentially the set of structures, systems, and components that are included in the Design Reliability Assurance Program, and I started to try to understand how that equipment was treated by going through Table 3.2-2 and I must admit I did not look at every line item in that Table, but, for example, the alternate AC gas turbine generators are assigned to Equipment Class V, which seemed consistent.

They're assigned to Quality Assurance

Classification A, which seemed consistent, and yet they're non-seismic, they're not assigned to Seismic Category II, so I was curious about that.

And then I started to think of other systems, and I used the main feed water system, condensate, and all of their support systems as an easy example, and none of that equipment is categorized as either Quality Assurance Classification A, Equipment Class V, or a Seismic Category II, yet the main feed water system is listed as a risk significant system in Table 17.4-1.

So I'm really curious about how that equipment was treated.

MR. HICKS: Risk significant equipment that's in the Table, 17.4-1, if it's not already safety-related in some way, I, II, or III, it should be classified as Equipment Class V.

CHAIRMAN STETKAR: Okay. You should take a look at that because none of, in Table 17.4-1, I mean it just may be just a typo for the alternate AC gas turbine as far as the seismic category because it is listed as non-seismic.

MR. HICKS: Well because it's Equipment Class V and risk significant doesn't necessarily require a seismic qualification. The seismic qualification is

driven by the Reg Guide 1.29, okay?

CHAIRMAN STETKAR: If you look at the third check mark on the slide that's up in front of us, it says Class V, Non-Safety-Related Components --

MR. HICKS: What that means is --

CHAIRMAN STETKAR: -- and the second sub-bullet says it's designed to meet the seismic requirements such as Seismic Category II.

MR. HICKS: What that means is that equipment that is designed to meet Seismic Category II is put into the Equipment Class V category, it doesn't necessarily mean that all Equipment Class V is Seismic Category II.

CHAIRMAN STETKAR: Oh, okay. I guess I -- I mean I read the, you know, the words in the report and --

MR. HICKS: Maybe this slide is confusing. This was intended to list the things that are included in Class V, not necessarily the reverse.

CHAIRMAN STETKAR: The reverse. Well, let me ask you then, why are the alternate AC gas turbine generators not Seismic Category II since one of the events that could give me a station blackout is an earthquake.

MR. HICKS: I think the simple answer is

that they don't fall within the Reg Guide requirements under 1.29 for being seismically qualified. I think they probably have, you want to answer the alternate AC, the specific design requirements for the alternate AC for seismic?

They're not classified as II or I because they don't meet the criteria to be II or I.

CHAIRMAN STETKAR: Okay.

MR. HICKS: And that's kind of the short answer I think. You want to elaborate on that all?

MR. NISHIO: No, we don't think we can call these a Fukushima related requirement. So the plant revision, plant specification we don't think that the Fukushima related requirement. No, we are discussing with the NRC.

CHAIRMAN STETKAR: Okay.

MR. HICKS: Fukushima related requirements are sort of outside of this.

CHAIRMAN STETKAR: Okay. So it may transition after the --

MR. HICKS: It may be one of those special systems that we apply seismic --

CHAIRMAN STETKAR: You know, I had Fukushima related issues in the back of my mind, but quite honestly, I tried to pick and choose a few of what I felt

were obvious examples and make sure I understood the classification scheme.

And, indeed, one of the first ones that I looked at were the alternate AC gas turbines and that got me thinking about should they be Seismic Category II and then I started to look for others.

The bigger concern in terms of the overall population though, to my mind, seems to be related to, not just the feed water system, because the problem is that Table 17.4-1 just has a single line item and it says "main feed water system."

It doesn't list pumps, it doesn't list, it just says "main feed water system" and it's classified as safety-related, or not safety-related, I'm sorry, risk significant, and in the Design Reliability Assurance Program.

Now if that's the case, and it's classified for the function of delivering water to the steam generators. So in order to accomplish that you need, not only the main feed water part of the system, you need the condensate system, you need the turbine cores cooling water system, you need the nonessential service water system, because all of those things support the main feed water pumps and the condensate pumps.

MR. HICKS: I guess it would depend on how,

what drove it to be put in the Table from the PRA and what assumptions --

CHAIRMAN STETKAR: It says delivering water to the --

MR. HICKS: Yes.

CHAIRMAN STETKAR: -- steam generators. So that means you have to get water through the pumps from point A to the steam generators.

MR. HICKS: I think what we'll have to do is look at that one.

CHAIRMAN STETKAR: Okay.

MR. HICKS: We'll have to look at that and maybe get back to you on --

CHAIRMAN STETKAR: Okay. I just thought maybe there was some general thing that I was missing.

MR. HICKS: I don't think so.

CHAIRMAN STETKAR: Okay.

MR. SPRENGEL: One clarification, the AACs will not change seismic classification.

CHAIRMAN STETKAR: They will not?

MR. SPRENGEL: So how we --

CHAIRMAN STETKAR: Okay.

MR. SPRENGEL: -- design, and treat, and evaluate those may be adjusted in terms of our Fukushima mitigation strategy.

CHAIRMAN STETKAR: But they will remain non-seismic?

MR. SPRENGEL: The classification themself, yes. So we're right now in discussion with the staff on additional commitments for the evaluation of those, but that does not necessarily mean it will change the seismic classification itself.

(Crosstalk)

CHAIRMAN STETKAR: There seems to be a fine point, but I guess we'll ask the staff about that.

MR. HICKS: Well I think it's kind of like some of these special systems that I listed earlier that we're going to apply some special design requirements to them, but we don't actually classify them as Seismic Category I or II.

MEMBER ARMIJO: The proposed special treatment, isn't de facto meet Seismic Category II?

MR. HICKS: I don't know how you interpret de facto, but --

MEMBER ARMIJO: Well I'm trying to say, you know, what's your objective as far as this equipment?

MR. HICKS: Of which? The AACs?

MEMBER ARMIJO: Yes.

MR. HICKS: The objective of the equipment?

MEMBER ARMIJO: Well, you know, in the

event of a severe seismic event that you want this equipment to continue to operate and function.

MR. SPRENGEL: I'm in the Fukushima analysis now and I --

MEMBER BROWN: Category II doesn't requirement them to continue to operate, just they're designed and constructed to maintain their structural integrity and then they're failure could degrade performance, but they're not required to remain functional and I mean --

MR. HICKS: And that's for a design basis events, yes.

MEMBER BROWN: Yes.

MR. HICKS: Right.

MEMBER BROWN: So I guess I was a little puzzled in response to the response to John's question on the AACs that, and I guess I forgotten how we covered the SBO circumstance. I don't remember --

MEMBER ARMIJO: I don't remember either, but --

MEMBER BROWN: -- we probably talked about that in one of the other meetings and I just don't remember.

MR. HICKS: They're not required to be seismic, so --

CHAIRMAN STETKAR: Well we know what they're there for, you know, their function, but this is the only time that we have actually seen pieces of equipment thrown into boxes, if you will.

MEMBER BROWN: Exactly.

CHAIRMAN STETKAR: That's fine. We've got an answer. We'll ask the --

MEMBER BROWN: Yes.

CHAIRMAN STETKAR: -- staff what they think now, but I certainly don't understand all of the nuances about, I understand the differentiation between Category I and II in terms of what a piece of equipment --

MEMBER BROWN: Yes.

CHAIRMAN STETKAR: -- is supposed to satisfy. It's my understanding that if something is non-seismic it basically can fall apart in a design basis earthquake.

MEMBER BROWN: Yes, if it's not required to remain functional following an SSC --

CHAIRMAN STETKAR: Right. And if it does fall apart it doesn't hit anything that's safety-related --

MEMBER BROWN: Exactly.

MALE PARTICIPANT: Yes.

CHAIRMAN STETKAR: Which is clearly true

for the alternate AC gas turbine generators based on where they're located they can't hit anything safety-related, but --

MEMBER BROWN: Yes. They maintain themself with integrity, which is key to that not becoming a missile.

CHAIRMAN STETKAR: But if they're non-seismic they don't even need to maintain their structural integrity.

MEMBER BROWN: That's, for Category II it does.

CHAIRMAN STETKAR: If they're non-seismic they don't even need to maintain their structural integrity.

MEMBER BROWN: Oh, okay, right. I'm sorry. Yes, I was in seismic --

CHAIRMAN STETKAR: And the alternate AC gas turbine generators are classified as non-seismic.

MEMBER BROWN: Non-seismic.

CHAIRMAN STETKAR: They can --

MEMBER BROWN: They're not Category II.

CHAIRMAN STETKAR: -- fall apart, in pieces, on the floor, and that's okay because, you know, according to the II versus I Category they meet the requirement that they will not impact a piece of

safety-related equipment if they fall apart in pieces on the floor.

MR. HICKS: As was noted, there may be some special design requirements though applied to the alternate AC generators --

CHAIRMAN STETKAR: Yes.

MR. HICKS: -- outside of the seismic --

CHAIRMAN STETKAR: And the nuances between why those special design requirements don't get a II in this particular box is beyond me. Apparently people who have law degrees are involved in that rather than engineers.

MR. SPRENGEL: I think it comes back to the perspective and I don't think we'd argue with a de facto Category II auto classification, but if we look at it of how we identify and put them into Seismic Cat I and Seismic Cat II, the AAC does not fit in Seismic Cat II.

So we agree with elevating our evaluation and testing and that, but we're not going to give it that classification because that's not what it's for.

CHAIRMAN STETKAR: Thank you.

MR. HICKS: Should we go to the next section or?

CHAIRMAN STETKAR: Anybody else have any questions about the categorization because we'll ask on

-- Pete?

MEMBER RICCARDELLA: I have a question about the seismic analyses in general. Are we going to cover more of that later?

MR. HICKS: Well the actual seismic analysis is in 3.7 and 3.8 and we're not covering those today. They're going to be covered in a separate meeting.

MEMBER RICCARDELLA: Okay.

MR. HICKS: Okay, we'll go to 3.11 then.

MR. GALVIN: This is Dennis Galvin. There will be a session on seismic qualification, but not seismic analysis. If you're question's on qualification --

MR. HICKS: Oh, right.

MEMBER RICCARDELLA: Just a general question, I mean, the NRC and the USGS, they're updating their seismic ground motion requirements in the U.S. Are you using these updated ground motion definitions?

MR. HICKS: I think the ground motion requirements are going to be in 3.7.1, right? And so I don't think we have anyone here that can really speak to it, but --

CHAIRMAN STETKAR: I think, Pete, the answer is we'll have to pick that up when we cover 3.7,

whether it's, it's certainly not covered in any of the sections that we're reviewing today.

MEMBER RICCARDELLA: Okay.

CHAIRMAN STETKAR: In terms of the ground motion spectra and whether it's a, I think it's 3.7? It's not a Chapter 2 issue --

MR. HICKS: No, 3.7.1 is the ground motion response spectra, I believe.

CHAIRMAN STETKAR: But today's session does cover whether something is I, II, or non-seismic in terms of Table 3.2.

MR. GALVIN: Yes, 3.2.

CHAIRMAN STETKAR: Yes. Thank you.

MR. HICKS: If there's no other questions then we'll got to, the next section is 3.11 that I'll cover and then we'll jump back to 3.3 when Mark Biery comes back up here.

Section 3.11 is the environmental qualification of mechanical and electrical equipment. This section describes the EQ Program. The Environmental Qualification Program demonstrates and documents in compliance with the requirements of 10 CFR Appendix A and general design criteria of IV and X, and 10 CFR 50.49.

The scope of the program, as described in

DCD Section 3.11, is consistent with the NRC approved guidance in SRP Section 3.11 and Reg Guide 1.206. One of the open items that we'll discuss later has to do with the scope description.

The program complies with applicable requirements delineated in Reg Guide 1.89 and environmental conditions which equipment qualification process addresses include the environment, seismic, chemical, radiation performance, and synergistic effects which has to do with system interactions.

Okay, in this section the term environmental qualification means the verification of design limited to demonstrating that mechanical, electrical, or I&C equipment are capable of performing their safety function under significant environmental stress that is in, you know, harsh environments resulting from design basis events in order to avoid a common cause failure.

For active mechanical equipment meeting the equipment scope definition and located in a harsh environment, compliance with the environmental design provisions of GDC-4 are achieved by demonstrating that the nonmetallic parts and components are suitable for postulated design basis environmental conditions.

For electrical and active mechanical

devices located in mild environments, compliance with the environmental design provisions of GDC-4 are achieved and demonstrated by proper incorporation of relevant environmental conditions into the design process, including equipment specifications.

And vendors will typically test this equipment using typically a nationally recognized testing agencies like Underwriters Lab and certify use in mild environments by using industry recognized ratings such as NEMA 1 or 3R.

The safety-related computer based I&C systems located in mild environments are qualified in accordance with Reg Guide 1.209. The general requirements for the environmental design and qualification can be summarized as follows.

Equipment is designed to have the capability of performing its design safety functions under all anticipated operational occurrences in normal accident and post-accident environments for the length of time for which the function is required.

The environmental qualification of equipment located in harsh environments is demonstrated by appropriate testing and/or a combination of testing and analysis.

And then the last component is a QA Program

that meets 10 CFR, Appendix B. Equipment qualified by the EQ Program is listed in DCD Table 3D-2, along with equipment addressed by the Equipment Qualification Program, that is, in other words, the Equipment Qualification Program is broader and includes things that are functionally qualified as described in Section 3.9 and also the seismic qualification requirements.

So the Equipment Qualification Program is the broad program and the environmental qualification piece is included in that. The equipment is identified by system, location, type, environment, and associated environmental parameters, and Appendix 3D provides a brief explanation as how this equipment and associated analysis was performed to establish the required environmental parameters.

For brevity, the DCD Table 3D-2 only identifies the associated equipment by device, tag, or instrument loop. However, the associate of the component sensor supports the mechanical piping valves, cables, penetrations, devices, and similar items needed for a fully functional device are also qualified in the EQ Program.

Technical Report MUAP-0805 is the equipment qualification of electrical mechanical equipment and it addresses the relevant environmental design and

qualification requirements of 10 CFR 5049 and 10 CFR Appendix A, GDCs, and Appendix B, the Quality Assurance Criteria.

With respect to systems and components being designed to withstand the effects of and being capable of performing their functions in the environmental conditions associated with the normal maintenance, testing, and accident conditions.

This report also addresses seismic qualification as described in Section 3.10 and functional qualification of active mechanical components as described in Section 3.9, which will be discussed later, and this is done as in integral US-APWR Equipment Qualification Program.

And the implementation of the Environmental Qualification Program is addressed, as I said, as part of that integral program. All right, there are eight open items in Section 3.11. This slide shows three in the first row and these, the first row has to do with the radiation dose and equipment locations assumed in the qualification process.

Changes were incorporated to address this item in DCD Rev 4 and these changes are under review by the staff. Second row item, has to do with equipment qualification data package template that has been

included in the MUAP.

The staff gave us some feedback on that template and MHI is currently revising the format of that template to address the NRC's comments.

These two items on this slide have to do with the scope of equipment addressed in the Program and the use of the term "important to safety" in DCD Section 3.11 and the MUAP Report.

MHI revised the scope description in DCD Section 3.11 to include text from the NRC approved guidance documents, and this item is still under discussion with the staff, between MHI's.

Next slide. The first item on this slide relates to ensuring that ITAAC adequately verify the EQ of electrical active mechanical components. Again, changes were made to the DCD ITAAC to address staff comments and the NRC staff is currently reviewing those changes.

The second item involves commercial grade dedication process. There was a comment about how that process was described in the MUAP and MHI is in the process of clarifying that description in the MUAP to address the comments provided by the staff.

And that concludes Section 3.11. Are there any questions? After this Section we're going to switch

players a little bit here, but we'll take any questions now for Section 3.11.

CHAIRMAN STETKAR: Questions?

MALE PARTICIPANT: Not me.

CHAIRMAN STETKAR: Switch players.

MR. HICKS: Thank you.

(Pause)

CHAIRMAN STETKAR: Mark?

MR. BIERY: Good morning, gentlemen. My name is Mark Biery, MNES, and I will be presenting the rest of the Chapter 3 sections covered today. Go on to the next slide, please.

We're going to first start with Section 3.3, Wind, Tornado, and Hurricane loadings. There are no open items for this DCD Section. Wind loadings, the design basis wind loadings are determined in accordance with the American Society of Civil Engineers and Structural Engineering Institute, made on design loads for buildings and other structures.

The US-APWR severe wind speed is 155 miles per hour with an annual probability of exceedance of 0.01. In this Table here we give tornado and hurricane loadings, including maximum wind speed, exceedance frequency, missile types evaluated, and applicable right guides.

CHAIRMAN STETKAR: Mark? Something that always is curious to me, and I really don't understand it, so perhaps you could explain it. And since we got through the first session in record speed we are now way ahead of schedule, so I can afford to ask you this.

MR. BIERY: Yes, sir?

CHAIRMAN STETKAR: The previous slide said that a 155-mile per hour design basis wind speed has an exceedance frequency of 0.01 per year, means it occurs once in 100 years.

This Table tells me that a 160-mile per hour happens to be associated with a hurricane, wind speed has a frequency of ten to the minus seven per year, once in ten million years.

For a 5-mile difference wind speed having five orders of magnitude difference in frequency to me just doesn't sound, I can't find the word, I will just say right. It does not sound right.

So could you explain to me how a 155-mile, 3 second peak gust wind speed at ten meters has a frequency of one event in a hundred years when I can then increase the wind speed and call it a hurricane and suddenly it only happens once in ten million years?

It's kind of hard to even measure a 3-second gust peak wind speeds within an accuracy of five miles

per hour depending on the measurement technology because there are certain different philosophies of how you do that, so I'm really curious about this five orders of magnitude difference in the frequency.

It comes into play, by the way, and I don't know whether, I didn't look forward, ahead in your slides, but, for example, you say that you design the turbine building such that the siding remains intact in a 155-mile per hour design basis wind, but it blows off in 160-mile per hour wind so that it does not, you don't have differential pressure during hurricane wind loading.

That's a pretty good structural design, too, to differentiate with a 5-mile per hour wind gust. So I'm really curious about this. And, by the way, I don't like to hear people just refer to I picked this number out of this book, because I can pick, I used to have a telephone book and I could pick any number out of that book that I could find.

So I'm interested in a fundamental, engineering perspective how this can be because it is your design basis.

MR. NAGAI: Okay.

CHAIRMAN STETKAR: You know, if you want to think about it and get back to us, you know, that's

perfectly fine. This is kind of a complicated question and answering it on the fly may be a little bit difficult.

But if you do want a caucus during a break or even get back to us tomorrow that's fine. I don't necessarily, this is a pretty difficult issue, so --

MR. NAGAI: We'll take an action --

CHAIRMAN STETKAR: Okay, thanks.

MR. NAGAI: -- and get back to you.

CHAIRMAN STETKAR: That's probably the best approach.

MR. NAGAI: Thank you.

MR. BIERY: Let us get back to you on that.

CHAIRMAN STETKAR: Yes.

MR. BIERY: The tornado and hurricane velocity pressure loads are computed in accordance with procedures accepted by SRP 3.3.2. No tornado wind speed adjustment applies for spectra height.

Tornado atmospheric pressure loading is computed for maximum pressure drop and there is no pressure drop effect evaluated for a hurricane. Load combinations are in accordance with procedures accepted by SRP 3.3.2 and are supplemented with the design criteria of procedures provided in Bechtel Topical Report, BC-TOP-3-A.

And again, there are no open items in this

DCD section, however, we will get back to you on your question though.

CHAIRMAN STETKAR: Yes, I'd appreciate that.

MR. BIERY: Sure.

CHAIRMAN STETKAR: Because it's, I'll ask the staff about it also because there are no open items on it, so that's kind of telegraphing, the staff may want to muster their forces for when they come up. Let me see my other notes, and another thing I did want to ask you regarding the, make sure I'm not ahead of myself, where are we?

Sometimes, you'll have to bear with me, I get lost. Okay. I was reading the section about the effects, 3.3.2.3, talks about the effects of failures of structures in components not designed for tornado and hurricane loads, and that's why I came across this statement that said "sections of the turbine building siding are designed," the entire turbine building is designed to withstand 155 miles per hour with no damage.

And then there's a statement that says "portions of the siding would be blown off in the event of a design basis tornado or a design basis hurricane," which means at 160 miles per hour your design to blow off sections of the siding to vent the turbine building so

that you don't have any differential pressures.

There is also a statement that says "any items, including the turbine building siding, which might become dislodged and become missiles under the maximum tornado or maximum hurricane conditions do not warrant further evaluation because they are considered to be enveloped by the missiles addressed in Subsection 3.5.1.4."

What I was mostly curious about is that the, if I go back to, again, this Table that I keep referring to, 17.4-1, for risk significant equipment, that Table lists the main transformers and the reserve auxiliary transformers as risk significant equipment, in particular, reserve all auxiliary transformers 3 and 4 because they're the normal supplies to the safety busses. Main transformers because they can affect overall power.

The question I had is, because you're not evaluating damage from wind damage to the turbine building siding and because the transformers are located adjacent to the turbine building, can the siding damage those transformers?

And I know generally where the transformer yard is located adjacent to the turbine building, and I was curious whether or not if the siding comes off you can short out all of those transformers.

And if you can, why it was deemed not necessary to evaluate the effects from that?

MR. BIERY: So if the siding is dislodged from the turbine building and it affects the transformers, did we evaluate it and --

CHAIRMAN STETKAR: Right. I mean, essentially, my understanding is that you determined that you didn't need to do an evaluation of any wind related damage to the turbine building, including the siding, because that damage would not affect any, you don't use the term safety-related equipment here, it's kind of vague, let me just call it important equipment.

And, as I said, the equipment that I could identify were those transformers because they are listed as risk significant back in that Table 17.4-1. And I was curious whether you thought about that and if you have an evaluation of why the siding can't damage the transformers, that's okay, that's good.

If it can damage the transformers I was curious why that's not important. And that's, again, something that you may want to take back.

MR. OGASAWARA: Yes. This is Ogasawara speaking. So wind and tornado design is a protective equipment. Equipment means structure on system on component. The target of the protective SSCs are just

Seismic Category I and the safety-related equipment.

So non-safety-related equipments and SSCs are not a target of tornado design except as a concept.

CHAIRMAN STETKAR: Okay. Okay, I'll ask the staff then why that's okay. Thank you.

MR. BIERY: The next section we'll discuss this morning is Section 3.5, Missile Protection. This section provides outline information on missile protection.

We define structure systems components be protected, credible missile selection, and design for protection from missiles, and we have two open items for this DCD Section.

Six types of missiles are considered for the US-APWR Missile Protection Design. Internally generated missiles, both inside and outside of containment, turbine missiles, missiles generated by tornados and hurricanes, site proximity missiles, except aircraft, and, of course, aircraft hazards.

Protection for internally generated missiles is provided. The COL applicant is to identify site specific SSCs to be protected from turbine missiles and assessed turbine generator orientation, including other units at multi-unit sites.

The design basis spectrum of tornado and

hurricane missiles conforms to that defined in their applicable regulatory guides. The applicant will verify site interface parameters with respect to aircraft crashes and transportation accidents.

CHAIRMAN STETKAR: Before we get to turbine missiles, because I'm going to have several questions there, there me ask you about several statements.

I have a long list of things here and rather than going through the long list let me preface it by saying the general concern. There are numerous statements throughout Section 3.5 regarding internally generated missiles.

They can be missiles from pipes, they can be missiles from pieces of equipment, pumps, turbines, gas turbine generators. It can be missiles from things being dropped, et cetera.

And there's a global statement in pretty much every one of those paragraphs that says "the probability of occurrence is therefore maintained less than ten to the minus seven," ten to the minus seven is a really, really small number.

If I look at things like, give you an example, in the main steam and feed water piping tunnels, you know, there's a statement that says it's less than ten to the minus seven per year, yet if I look at the PRA,

frequencies of main steam line breaks outside the containment are on the order of once in a hundred years.

Frequencies in main feed water line breaks outside the containment are on the order of three times in a thousand years, three times ten to the minus three. So I was curious what type of analyses you did to justify the fact that you're claiming that the frequencies of missiles from these breaks is less than ten to the minus seven per year.

Statements saying probability of missiles from a gas turbine generator is less than ten to the minus seven per year. Where are the analyses that you performed to justify those exceedingly low frequencies? They are extremely small.

MR. BIERY: So with regard to the global statement?

CHAIRMAN STETKAR: I mean there are statements throughout this document where you do systematically go through a long list of potential missile sources, as I said, anywhere from piping generated missiles to locating equipment generated missiles to drop type missiles.

And in each of those sections the missile risk or threat, or however you want to characterize it is uniformly dismissed with a simple statement saying the

probability is less than ten to the minus seven.

And I can accept that if I can see an analysis to justify why that's so. I cannot accept it on face value with just saying it's less than ten to the minus seven, because ten to the minus seven is a very, very, very small number.

So to justify that somebody must have done an analysis. You can't just say it. I mean you might as well say it's less than ten to the minus 100 if you're going to say a number.

MR. OGASAWARA: Well my understanding is that we have no, those numbers are coming from the practice, general practice barriers.

CHAIRMAN STETKAR: General Practice? So we have ten million years of operation of this equipment without any generation of a missile? We don't.

One example is, I personally, I have stood next to the chunk of steel that was thrown through the casing of a circulating water pump at an operating nuclear power plant which shall renamed unnamed, that went through the casing, through the wall of the room, and went out skidding and embedded itself in the ground several meters from the outside of the building, fortunately missing the alternate AC gas turbine generator which was a few feet from where it landed.

I have seen this. Since I have seen it, it is probably more frequent than ten to the minus seven per year and yet there's a statement saying "missiles from non-safety-related rotating equipment is less than ten to the minus seven per year.

Now maybe I was unlucky, you know --

MEMBER ARMIJO: Or you're very, very old.

CHAIRMAN STETKAR: I'm feeling older every day. But that's, you see my sense here, is if you're making frequency arguments to justify dismissing something --

MALE PARTICIPANT: You better --

CHAIRMAN STETKAR: -- I'd really like to see your technical justification for those frequency arguments. If you're making qualitative arguments based on other considerations, then just admit they're qualitative arguments based on other considerations.

But as soon as you start putting those numbers to things, I'd really like to see justification for those numbers.

MR. BIERY: Sure.

MR. OGASAWARA: Yes, ours was a justification, basically as is described in DCD's Chapter 3.5, pipings and valves, those are designed in accordance with ASME Code, Section III and our conclusion

of our DCD description is based on the, the design is based on the ASME so those probabilities are less than ten to the minus seven.

CHAIRMAN STETKAR: I, you know, I read that, and if that's the case since the main steam and main feed water piping are also designed to those same codes, how come their break frequencies are on the order of once in a hundred years?

Why is the large LOCA frequency much larger than ten to the minus seven per year? Simply by saying that something is designed to a certain set of standards does not necessarily guarantee that the frequency is less than ten to the minus seven per year.

And that's my whole point. I mean the claim in general is that if something is designed to the standards, essentially you're saying it won't break and yet we have experienced that things designed according to standards occasionally do break.

MR. SPRENGEL: Okay, well --

CHAIRMAN STETKAR: Not very frequently, but you get my point. I mean --

MR. SPRENGEL: Yes. The important thing is not focusing on one or the other, it's understanding why there's differences in terms of how we're looking at that frequency or probability.

CHAIRMAN STETKAR: That's right. It's, that's right. My only concern, I have to be careful in terms of making my own conclusions because I indeed read about each of those things and I have questions about each one individually.

They tend to focus on this ten to the minus seven number, but for other qualitative reasons for particular pieces of equipment they sounded like a reasonable qualitative basis.

But as I said, as soon as you start relying on a number there ought to be justification for that and because we have evidence that things do indeed fail, despite the fact that they're designed to codes and standards, you need to address that if you're going to rely on those numbers.

MR. SPRENGEL: Okay. We'll follow up on that item.

CHAIRMAN STETKAR: Thanks. Now we can talk about turbine missiles.

MR. OGASAWARA: And also, I will check up, basis of the probability of Chapter 19 because Chapter 19 basis may be different from Chapter 3, so I will check that point.

You mentioned the probability of Chapter 19 --

CHAIRMAN STETKAR: Right. Chapter 19 is the PRA and Chapter 19 is supposed to be a best estimate analysis for evaluating, initiating event frequencies and that best estimate analysis in many cases is informed either by actual experience data, in other words, counting up failures and dividing by a number of exposure years, or in some cases like large LOCA frequency, it's based on analyses that people have done, because we haven't had any large LOCAs.

And, you know, as I mentioned, those frequencies, pipe break frequencies in particular, are much higher than ten to the minus seven. Now not necessarily every pipe break will generate a damaging missile, but a large pipe break will probably, of a high energy line, will probably generate some amount of missiles.

How large they are? How energetic? You know, that might be part of your analysis if you had an analysis. But other statements, as I said, you know, just a simple statement that says non-safety-related equipment, rotating equipment, cannot generate missiles at a frequency higher than ten to the minus seven per year, maybe I was not lucky then.

MR. BIERY: I think we understand your concern and we'll definitely look into that for you.

CHAIRMAN STETKAR: Yes.

MR. BIERY: We'll now move on to turbine missiles. Two categories of turbine failures are evaluated. The first, design over-speed failure associated with brittle fracture as well as destructive over-speed failures associated with ductile failure.

And two supporting technical reports have been submitted by MHI, the first is probability of missile generation from low-pressure turbines and the second is probabilistic evaluation of turbine valve test frequency.

The design basis spectrum of tornado and hurricane missiles --

MALE PARTICIPANT: Sorry.

MR. BIERY: Oh, I'm sorry. That's better. Turbine missile generation probabilities, particularly from a brittle fracture within the range less than design over-speed is less than ten to the negative five per year, unless the in-service inspection intervals exceed 20 years.

And destructive over-speed, less than ten to the negative five per year by quarterly turbine valve test frequency.

MEMBER BROWN: In other words you test it every quarter? Is that what that means, you're going to

be running a test every quarter on over-speed? In which, you say the words "less than ten to the minus five quarterly turbine valve test frequency," and I guess I don't understand the less than ten to the minus fifth, that's consistent with Chapter 10 under your turbine rotor integrity discussion.

But you through in this bi-quarterly, but I don't understand what that means.

MR. BIERY: So the quarterly turbine valve test frequency is your question? It will be done quarterly.

MEMBER BROWN: Well, I mean, how that's related to the, if the destructive over-speed is less than ten to the minus fifth per year, bi-quarterly turbine valve test.

MR. SPRENGEL: Yes. The frequency of the testing feeds into the probability evaluation.

MEMBER BROWN: So that's part of the analysis calculation? I mean it's part of the --

MR. SPRENGEL: Yes.

MEMBER BROWN: -- it's an input into the calculation?

MR. SPRENGEL: Yes. And so we're just qualifying that because that's what's there for the design certification. You know, off in to the future

specific plans could change their test frequency based on other plant conditions and redo their evaluation.

MEMBER BROWN: But that assumes that somebody's run some, I mean if you're doing it by some test frequency, I'd assume somebody's run some tests, I presume, or is this just an analytical construct based on the design of the turbine, and in other words how do you arrive at that?

MR. SPRENGEL: Currently it's only analytical, yes, and that feeds into the report material in Chapter 10.

MR. BIERY: SRP 3.5.1.3 prescribes that the P1 of the favorably oriented turbine to be less than ten to the negative four and that gives us a P4 with an acceptable risk screen of ten to the negative seven per year.

The US-APWR turbine is favorably oriented with the turbine access aimed at the reactor building. The reactor building, the containment, and the power source building and safety-related and non-safety-related SSCs within these structures are located such that the turbine generator is favorably oriented.

And P1 for the US-APWR turbine is maintained to be less than ten to the negative five by using a proper

rotor design material selection, pre-service and in-service inspection programs, and redundant protection and control system, and we give these details in Chapter 10.

CHAIRMAN STETKAR: Okay. Are you going to be addressing this discussion, the effects of turbine missiles on the other unit or not?

MR. BIERY: I believe that's the COL, response to --

CHAIRMAN STETKAR: Okay, that's the COL. I'll ask the COL when we come up to that one. I read through MUAP-07029 and I understand how the analyses in that document were performed.

I have numerous questions about the analyses in that document. The first question that I have, and it's actually a statement, those analyses are not complete.

Those analyses evaluate only failures of the main turbine stop valves, the main turbine control valves, the reheat stop valves, and the intercept valves.

They do not evaluate failures in the hydraulic system. They do not evaluate failures in the over-speed protection control system, including the over-speed instrumentation. They are simply not included.

So the first observation is that the turbine over-speed frequency is underestimated. By how much? I'm not going to speculate on that because that's not my job to do. I could speculate. I have equations. I did a little analysis, I can make up some numbers, but the analyses are not complete.

This is a typical problem, because what you've done is you've had someone look at the turbine as if it's in isolation with the rest of the world and ignored everything in between.

So I would recommend that you go back and somehow, somewhere, account for the, everything from the speed pickups on the shaft of the turbine all the way through the electronics that processes those signals through the solenoid valves that have to dump hydraulic fluid and the valves that you've evaluated so far.

And I don't know, in the interest of time, I have a lot of questions about how you derived your failure frequencies for the turbine stop valves and turbine control valves, because in many cases you didn't have any operating experience.

And in those cases it wasn't at all clear to me how you derived the failure rate for the piece of equipment based on no operating experience. I think that I understand how you did it, but it's not described

very well in the report.

So I'd like a little bit better information about how you derived those failure rates for pieces of equipment with no actual observed failures.

You do, the report does list very well as a matter of fact, better than most people do, the data that you used, the operating experience data from the Japanese units that you used to develop the failure rates.

MR. BIERY: Sure.

CHAIRMAN STETKAR: And as I said, the cases, and I understand where you did have failures, I reproduced the failure rates and I understand how you did that. I don't understand how you did it for the cases where you had zero events.

And that could be important because a lot of the equipment had zero events.

MR. BIERY: Okay.

CHAIRMAN STETKAR: So that's also a general comment. And I think I'll leave it at that. The others are way too detailed for this forum.

MR. BIERY: So to --

CHAIRMAN STETKAR: So anyway, to recap, at least at this stage, what I'm interested in is for the equipment that you did model, the valves basically, how

you derived the failure rates, the basis for the failure rates that you used for the valve types for which there were no observed failures in your experience database. So that's one issue.

And then the second issue is just this general statement that there's the equipment from the speed pickups through the valves that dump the hydraulic fluid that's basically not considered in the model.

MR. BIERY: So your overall concern is just --

(Crosstalk)

CHAIRMAN STETKAR: The concern is, and I don't know how much that middle equipment would contribute to your estimated over-speed frequencies here.

MR. BIERY: I think we'll look into that, sir.

CHAIRMAN STETKAR: Okay. Thank you.

MR. MINAMI: I'm sorry. I have one comment. At the beginning of your question you said that our technical report is incomplete and do not deal with the probability of the failure of control and protection system, right?

CHAIRMAN STETKAR: Yes.

MR. MINAMI: And we provided two technical

reports. The first one is the report dealing with the probability of the turbine rotor failure, turbine rotor failure.

CHAIRMAN STETKAR: 028.

MR. MINAMI: Right.

CHAIRMAN STETKAR: That's correct.

MR. MINAMI: And the other one is analysis.

CHAIRMAN STETKAR: 029.

MR. MINAMI: Right, 29, and then dealing with the probability of the failure of protection and control system. And I don't understand why our report is incomplete? Because we provided it to --

CHAIRMAN STETKAR: Okay, 07029. 07028 deals with things that I don't personally understand that other of my colleagues do understand and that is mainly materials types issues, crack growth --

MR. MINAMI: Right.

CHAIRMAN STETKAR: -- crack propagation, it's a mechanical failure of the turbine under different types of load cycling conditions under different over-speed conditions.

Normal speed, 110 percent over-speed, 120 percent over-speed, okay, that's what I would characterize as mostly a materials oriented report. The second report, MUAP-07029, is the report that estimates

the frequency at which an over-speed will occur, at which the over-speed protection system will not trip the turbine before an over-speed occurs.

The second report is the one that I'm taking issue with and that report simply evaluates as best as I can tell, now maybe I'm missing something, but I think that I reproduced all of the equations, evaluates only failures of the main turbine stop valves, the main turbine control valves, the intercept valves, and the reheat stop valves, and their associated logic.

And the data, I understand the data that's developed for the valves is developed on a valve type by valve type basis. That study does not evaluate any contribution from the instrumentation and control system or the hydraulic system from the instrument pickups on the turbine shaft all the way through the over-speed protection control logic, solid state stuff, you know, the digital, all the way through the solenoid valves that eventually drop fluid.

That system, that end-to-end part of this system from the speed pickups through the valves that drop the fluid pressure, is not part of the turbine control valve and stop valve. Data or model?

So what's missing from the MUAP-07029 analyses are an evaluation of how frequently, what's the

contribution from failure of everything from the speed pickups up to and including the solenoid valves that dump hydraulic fluid.

That's just simply missing. It would add to the frequency that's calculated from simply failures of the end use devices which are the stop valves and control valves themselves.

And you can't argue that it's included in those stop valves and control valves because in my experience the designs of the electronic systems and the hydraulic systems differ from plant to plant.

They may be standardized in Japanese plants, but they also have common cause failures and in a sense that failures of, I think, I'd have to look up equations, I think it would require failures of two solenoid valves, but if those two solenoid valves fault, none of the turbine stop valves and control valves would close.

In other words fluid would not be drained. So they're not something that you can say, well failure of the solenoid valve was somehow included in the stop valve or control valve because the functional effects are different.

So that's the report, to answer your question, that MUAP-07029 is the report that I'm taking

issue with in terms of completeness of the analyses.

MR. MINAMI: I understand what you said.

CHAIRMAN STETKAR: Okay. Thanks.

MR. MINAMI: But when I investigate our experience in Japanese PWR plants and we investigated every failure of the turbine control system, which include the valve and the actuator, valve or sensing around control systems.

And if the, I'm not sure at this moment, but if some solenoid failed to function the turbine should be tripped, right. And if the turbine tripped occurred, this event, this included in this analysis.

So my intention is that I didn't explicitly show the incident of the, for example, for the pickup or for the micro-CPU or for the cable or something like that, not divided into the detail, but I included all the component included in the turbine control and protection system.

And if any failure in that system occurred, this incident is counted in the Table and MUAP-07029. And I think it, for me, for us, it's very difficult to find out which component failed or which component did not function during the operation.

So I believe, in my sense, every component failure have been incorporated in our analysis. So this

is my understanding --

CHAIRMAN STETKAR: I hear what you're saying. I took some time to look at the design information about the over-speed protection circuits. I took some time to look at the design of the hydraulic fluid circuits and I took some time to write up, I didn't actually develop fault trees, but I wrote up some algebraic equations that has and or logic in it to at least describe my understanding of the logical combinations of failures that could result in a failure to trip, and I came up with a large number of potential failures.

In particular, if I look at the hydraulic system there are one, two, three, four, five, six, seven, eight combinations of two solenoid failures, two solenoid valve failures, that could result in failure.

Now your experience base may never show a failure of the turbine to trip because failure of only one of those solenoid valves would not cause a failure of the turbine to trip. You would need failure of at least two.

MR. MINAMI: Right.

CHAIRMAN STETKAR: Okay. On the other hand, you've never had an over-speed failure either. You've only had single failures of turbine stop valves

and control valves.

You never had two of them fail, at least in my understanding looking at your data. Ten to the minus five per year is a really, really small event. You would not have expected to see any of these events in the entire operating history of all the Japanese nuclear power plants.

You would require many more hundreds of years of operation to even get into that range. So just by saying we haven't seen it doesn't mean that that justifies that it's smaller than ten to the minus five.

I mean that's basically why we do risk assessments, basically why we develop these models.

MEMBER SCHULTZ: It doesn't mean that that shouldn't be modeled in the overall evaluation --

CHAIRMAN STETKAR: That's right.

MEMBER SCHULTZ: -- and when you're looking at how test frequency would impact the system.

CHAIRMAN STETKAR: Right. If, for example, the analysis did include, or does include, and I missed it somehow, the solenoid valves in the hydraulic unit in the, you know, hydraulic fluid system, if for some reason their failure rate is such that it would justify a more frequent testing, you know, the analysis should identify that.

Because those solenoid valves would also be tested by, you know, the operation would be tested by some of the tests that are performed.

MR. MINAMI: Yes.

CHAIRMAN STETKAR: Anyway, I'll just leave it at the table, we've spent a reasonable amount of time. If you can show me where, you know, where those things are included or where, either in the model, and I don't think they're included in the model, or in the data, I'd be happy.

I just don't think they're there because I used your data and I used your model and I pretty much reproduced your results, so I --

MR. MINAMI: Yes.

CHAIRMAN STETKAR: I'm curious.

MR. MINAMI: Yes, I understand, yes. In our system, one single failure doesn't lead to the triggers, but --

CHAIRMAN STETKAR: There are no single failures anywhere that I could find and it's obvious.

MR. MINAMI: Right, right. It's all right, but we design so that single failure did not lead to --

CHAIRMAN STETKAR: True.

MR. MINAMI: -- the turbine trip, so --

CHAIRMAN STETKAR: And everything that I could find in things that I looked at, end to end, confirmed that. There is, I could not find a single failure.

MR. MINAMI: So what --

CHAIRMAN STETKAR: I can find common cause failures of two solenoid valves, I can find large numbers, as I said, at least eight combinations of two failures of valves --

MR. MINAMI: Right.

CHAIRMAN STETKAR: -- in addition to the ones that you evaluated that can lead to failure.

MR. MINAMI: So what I want to say here is that a single failure of a component cannot lead to the turbine trip and this event did not jeopardize the turbine system, main turbine system.

So this is a purpose, why we design the component to be turned to turbine type divergent or something like that. So I think we do not need to consider single failure on a single failure into this analysis.

It's my understanding why to, I'm not sure at this moment.

CHAIRMAN STETKAR: Let me just reiterate. In the investigations, let me call it that, that I

performed, I couldn't, I agree with you completely. I could not find any single failures --

MR. MINAMI: Right.

CHAIRMAN STETKAR: -- that would result in failure of the turbine to trip, but that's, and that's good. I mean if I had found a single failure I would have phrased my initial concern a lot differently.

I could not find a single failure. However, the purpose of the analyses in MUAP-07029, one of those purposes is to examine whether or not there are single failures.

The other purpose is to identify an appropriate testing frequency for that integrated system to give you assurance that the failure rate of the turbine over-speed protection system, integrated failure rate end-to-end, remains low enough so that your turbine missile frequency, P1, is less than ten to minus five per year.

And that testing frequency is determined by the integrated reliability from the speed sensors all the way out through the actuated valves, and if there's something missing in that and if the missing equipment is more important than the actuated valves, then your testing frequency may be, or the interval, may be too long.

Because the way the failure rates are derived, they're integrally related, linerally related to the testing interval.

MR. NAGAI: We will go back and take a look at details and I think after we have --

CHAIRMAN STETKAR: I have my own, as I said, I don't want to speculate about how important the equipment in between may be, I have my own opinions, but that's just based on numbers I have and it's not worthwhile to speculate here.

It's more important to understand whether or not the analyses are complete.

MR. MINAMI: Let me have one comment.

CHAIRMAN STETKAR: That's fine.

MR. MINAMI: In addition to the quarterly valve test we usually do the component test, online test, for example, solenoid valve is tested once a month, right.

And if we carry out such kind of component testing on the line we can assure that this component is still all right or not. So if we consider such facts into the analysis or assessment, at this moment I am not sure, but I don't think we need to consider the single failure into this analysis, but that we would consider --

CHAIRMAN STETKAR: Okay. I mean monthly

testing, it's basically a question about how many failures of each component have you seen given the typical testing interval for those components because that's the way the failure rates are derived.

So if you do monthly testing of the solenoid valves, how many failures of those solenoid values have you seen given that monthly testing interval and that would then, that failure rate would then be reflected in your model to generate the over-speed protection, the integrated over-speed protection failure.

MR. MINAMI: I'm not sure I can get such kind of data, but I will --

CHAIRMAN STETKAR: Well that's okay. The second question addresses the issue of you had failure rates for pieces of equipment that you have zero observed failures, so, you know, not having data available didn't stop you from estimating failure rates for certain types of valves.

I think we've probably discussed this enough, but --

MR. MINAMI: Okay.

CHAIRMAN STETKAR: We can have more discussions offline if you want to discuss some details.

MR. MINAMI: Right.

MR. BIERY: Okay. The structure systems

and components are protected from external missiles by the reinforced concrete external walls and roof of the safety related reactor building and power source building.

Components, protective shields, and missile barriers are designed to prevent damage to safety-related components by absorbing and withstanding missile impact loads.

A target SSC shields and barriers are evaluated for both local effects and overall structural effects through these impacts. And again, we have two open items for this section. The first, MHI is requested to provide an analysis assessing the local effects of an automobile missile in all Seismic Cat I structures.

A response was submitted in April and the staff is reviewing this response. The second open item is with regard to ITAAC. The first concern being key design features, information regarding the turbine orientation being included in DCD Tier 1.

MHI went ahead and revised Tier 1 to include the description of the turbine being favorably oriented with respect to the nuclear island. The second concern was the acceptance criteria of the ITAAC Table stating that the turbine missile probability analysis exists and conclude the probability is less than ten to the negative

five per year.

And MHI went ahead and revised the acceptance criteria and the RAI associated with that concern is under review by the staff.

MEMBER RICCARDELLA: Is the staff reviewing your document that performs the analysis to demonstrate this ten to the minus fifth probability?

MR. BIERY: I missed a part of --

MR. SPRENGEL: The answer is yes. Yes, as part of the Chapter 10 review primarily.

MEMBER RICCARDELLA: Okay.

CHAIRMAN STETKAR: By the way these are organized, Pete, because you haven't been involved in a lot of this review, MHI or MNES collectively develops, there's two types of reports that they develop.

One is a topical report which the staff reviews and writes a separate safety evaluation for, for example, the advanced accumulators. They also produce many technical reports that provide details of analyses --

MEMBER RICCARDELLA: Right.

CHAIRMAN STETKAR: -- which are then summarized in the formal design control document.

MEMBER RICCARDELLA: Those are the --

CHAIRMAN STETKAR: The two that we were

talking about, the 028 and 029 are technical reports, the staff reviews those as part of their review of the section of the design certification documents.

MEMBER RICCARDELLA: Okay.

CHAIRMAN STETKAR: So the staff should have reviewed those reports as part of this.

MEMBER RICCARDELLA: Yes, I've looked at the 028 Report --

CHAIRMAN STETKAR: Yes.

MEMBER RICCARDELLA: -- and I have questions on it --

CHAIRMAN STETKAR: Okay.

MEMBER RICCARDELLA: -- and I'm just not sure how to get those --

CHAIRMAN STETKAR: If they're questions regarding the actual analysis, it's time to ask them to MHI right now because that's what we're talking about.

MEMBER RICCARDELLA: And I'm looking at the analyses that predicted the probability of failure due to low-cycle fatigue in the 028 document and I really don't see how you've gotten to the probability of failure.

I see, you know, some general statements about the variability, for example, of the NNC parameters in the Paris equation and in the very next table is

probability of rupture due to low-cycle fatigue and I just can't see that the chain, how you got to there.

It's just not described very well, but it might be something that I could discuss with the staff.

CHAIRMAN STETKAR: Either way. I mean if, MHI did the analyses --

MEMBER RICCARDELLA: Yes.

CHAIRMAN STETKAR: -- so they should have an answer for how the values were developed.

(Off the record comments)

MR. MINAMI: Just a moment, please.

CHAIRMAN STETKAR: If it takes some time to find that answer, you know, you may want to wait and get back. Do you understand the question based -- okay.

MEMBER BROWN: There's another --

CHAIRMAN STETKAR: Before we ask a --

MEMBER BROWN: Oh, I'm sorry.

CHAIRMAN STETKAR: -- separate question, let's see if --

MEMBER BROWN: Well it's not separate, it's related, so go ahead.

MR. NAGAI: There's a separate slide that we prepared --

CHAIRMAN STETKAR: Okay.

MR. NAGAI: -- that may answer his

question.

CHAIRMAN STETKAR: Okay.

MR. NAGAI: I'm not sure if it is appropriate timing to --

CHAIRMAN STETKAR: Let me ask you this, we're getting close to time for a break. If you want to discuss this during the break we can come back to it after the break.

MR. NAGAI: Okay.

CHAIRMAN STETKAR: It'll give you a little bit of time to decide, you know, if you do have a slide that's been prepared to address it that would be great.

In the interest of completeness since Charlie did say that he had a question related to this general topic --

MEMBER BROWN: It's not, no, to help, there's another topical report on probability of missile generation from low pressure turbines, is that related to this, where they talk about low-cycle SSCs --

CHAIRMAN STETKAR: Just to make things clear, that we're all communicating, you're talking about Report MUAP-07028-P, correct?

MEMBER BROWN: Yes, I've got that.

CHAIRMAN STETKAR: And you're talking about the same report?

MEMBER BROWN: No, 10005-P, and it's a probability of missile generation from low pressure turbines and there is some discussion in there on the probability of low pressure turbine brittle rupture due to low-cycle fatigue during startup, shutdown cycles, et cetera.

Now I didn't know if that was related, I wasn't sure whether you had that report, it's one I've had for three years.

CHAIRMAN STETKAR: What's the date on that? I don't even --

MEMBER BROWN: REV 0 is what I've got.

CHAIRMAN STETKAR: What's the report number, Charlie?

MEMBER BROWN: It's MUAP-10005-P.

CHAIRMAN STETKAR: Huh? I don't happen to have that report so I'm, I haven't even looked at it.

MEMBER BROWN: And it's Probability of Missile Generation From Low Pressure Turbines it's called.

CHAIRMAN STETKAR: I wonder if that has been superseded by, REV 2 of MUAP-07028-P and the title of that report is, curiously, Probability of Missile Generation From Low Pressure Turbines --

MEMBER BROWN: Yes.

CHAIRMAN STETKAR: -- is dated June 2013.

MEMBER BROWN: Yes. This one I got back in July 25th roughly of 2011, so it's the report I got.

MR. MINAMI: Excuse me. I have answer. We have two technical reports regarding the one low pressure turbine missile analysis --

CHAIRMAN STETKAR: Okay.

MR. MINAMI: -- and the first one is analysis for the low pressure turbine rotor with 74-inch class rotating blade and --

CHAIRMAN STETKAR: Okay. But --

MR. MINAMI: -- a 74-inch class rotating blade. And the 10005 is an analysis for the low pressure turbine with 54-inch class rotating blade. So that --

MEMBER BROWN: Okay.

CHAIRMAN STETKAR: So it's for a different, it's a different turbine design then?

MR. MINAMI: The design is different.

CHAIRMAN STETKAR: If the one that's referred to in the DCD is --

MR. MINAMI: 07028 --

CHAIRMAN STETKAR: -- 07028, right. Okay. So that's the one that you --

MEMBER RICCARDELLA: That's the one I have a question --

CHAIRMAN STETKAR: Yes.

MEMBER RICCARDELLA: Okay.

MEMBER BROWN: Don't ask me why I have it, it just, it was sent to me, and goes into the topical report file.

CHAIRMAN STETKAR: Technical report.

MEMBER BROWN: Technical report file in this case, yes.

CHAIRMAN STETKAR: Okay. So for --

MEMBER BROWN: Sorry for the interruption.

CHAIRMAN STETKAR: That's fine because I didn't even see that one. So anyway, let's do this, let's take a break and let MHI caucus among themselves to see if they can provide a little bit more elaboration on the low-cycle fatigue failure probability to answer Pete's question.

MEMBER RICCARDELLA: Yes. Just to --

CHAIRMAN STETKAR: Do you have any other ones that you want to get on the table, too?

MEMBER RICCARDELLA: Well, no, my question is specifically about Section 3.3 of the 07028 Report.

CHAIRMAN STETKAR: Okay.

MR. MINAMI: Yes.

CHAIRMAN STETKAR: Good. Let us take a break and because we're a little ahead of schedule and

because --

MR. OGASAWARA: I think --

CHAIRMAN STETKAR: I'm sorry, go on.

MR. OGASAWARA: Before going to the question, can I answer to the question in Section 3.3 the wind speed 155 and also 160 questions?

CHAIRMAN STETKAR: Sure.

MR. OGASAWARA: Could you show the presentation, Page 30? One hundred fifty-five is the probability of ten to the minus two per year. This is a severe wind and 160 miles per hour is a ten to the minus six, that is extreme wind and in our DCD we call it as hurricane.

And those numbers are not related to each other and those are defined independently. So if we calculate and come up a number of 155 that probability of ten to the minus two, to the probability of a ten to the minus six, the number would not be 160 miles per hour.

But severe wind and extreme wind, those numbers are calculated base on a calculator and those numbers different rotor combinations calculation of missile, and those rotor combinations are shown in Chapter 3.8.

And the coefficient provide to the severe wind and also the coefficient numbers to the extreme

winds are different. And also we have the core items for severe wind and also extreme wind.

If in the site conditions the winds are higher than 155 or the number is higher than 160, the COL applicant have to show, has the responsibility for the design.

CHAIRMAN STETKAR: I understand the, I certainly understand the second part. The first part, I'm still, both of those wind speeds, to me, and I'm not a structural engineer, but both of those wind speeds are characterized as a 10-meter, 3-second gust wind speed if I read the report.

And I understand that, well I'm not sure that I understand why an extreme wind of 160 miles per hour that's called a hurricane should be considered differently from a severe wind of 155 miles per hour which is called a severe wind.

Because if they're both 3-second straight lined wind speeds, a hurricane doesn't have any rotational force. I don't know why they're treated differently because, as I said, I'm not a structural engineer.

My basic curiosity is why a 5-mile per hour difference in wind speed results in five orders of magnitude difference in the estimated recurrence

interval or frequency. To me that is curious.

MR. SPRENGEL: We'll take some more time --

CHAIRMAN STETKAR: Yes.

MR. SPRENGEL: -- and come back to you on that.

CHAIRMAN STETKAR: Thanks.

MR. SPRENGEL: We can even --

CHAIRMAN STETKAR: Thank you.

MR. SPRENGEL: Sure.

CHAIRMAN STETKAR: Anything else? Let's take a break. I actually need to take a break, I was trying to get to this, and let's make it for 20 minutes, let's reconvene at 10:35 a.m.

(Whereupon, the foregoing matter went off the record at 10:16 a.m. and went back on the record at 10:36 p.m.)

CHAIRMAN STETKAR: Back in session. Mark, it's yours.

MR. BIERY: Yes, sir. Up front I'm now joined by Masashi Ito, we shuffled seats again while we were on break. As far as the outstanding questions are concerned, we're going to go ahead and we're going to answer some of those after lunch.

CHAIRMAN STETKAR: Good.

MR. BIERY: So, for now, we're just going

to push on to Section 3.4.

CHAIRMAN STETKAR: That's fine, you know, we've done this a lot and that seems to be a good model. If you can get back to us with some answers sometime in the next, you know, before the end of the day tomorrow, that would great.

And as always, MHI has always been very, very good on following up on questions, so it's mostly for us to get the question out on the table, make sure that we understand the question mutually and then get back to us when you can. Thank you.

MR. BIERY: Sure. With that we move on to Section 3.4, Water Level Design. US-APWR is designed to accommodate the effects of external or internal flooding.

Protection of plant nuclear safety functions during and after internal and external flooding events is addressed in Subsection 3.4.1. Design conditions for Seismic Cat I structures to withstand a hydrostatic/dynamic loads from design basis flood or ground water conditions is addressed in Subsection 3.4.2.

External flood protection, water sources that are considered are listed here, in particular, probable maximum precipitation and ground water.

Design basis flooding level for US-APWR Standard Design is one foot below the plant grade and site specific external flood conditions will be addressed by the COL applicants.

Key design features include thick reinforced concrete external walls and base mats, penetration with flood protection features as well as sloped roofs with drainage systems.

Site specific design features, again, addressed by COL applicants. Internal flood protection, water sources considered include earthquakes, which is failure of non-seismic components, pipe breaks and cracks, in accordance with Section 3.6.2, firefighting ops, and pump mechanical seal failures.

Key design features for internal flood protection include physical separation of safety-related SSCs by protective barriers, watertight doors, and penetration seals to preclude simultaneous loss of redundant systems.

Enhanced piping design is used to minimize postulated flooding water sources and placement of safety-related SSCs above internal flood levels is used. The NRC staff conducted an audit of these design documents in this past April.

And office procedures, these discuss

flooding effects to Seismic Cat I structure design and Section 3.8 provides the design and analysis procedures used to transform the static and dynamic effects of the design basis flood level and ground water levels applied to these Seismic Cat I structures.

Again, Section 3.7 and 3.8 we're going to discuss at a later date. We have some open items for this DCD Section. The first, this RAI asked MHI to provide the basis for assumptions made for evaluating flooding due to the firefighting operations and how to discuss high and moderate energy line breaks and cracks being accounted for in the flooding analyses.

An RAI response was submitted in July 2013 after the technical audit this past April, and the staff is evaluating this response. Updates and changes have been incorporated into DCD Rev 4 to reflect this.

The next open item, MHI was asked to update Figure 3K-5 to show flood barriers at the main control room vestibule. A response was submitted in October of 2012 stating that the figure would be updated in accordance to the closure plan for seismic and structural analyses. The updated figure is also in DCD Rev 4.

The next open item, MHI requested to further clarify the use of 0.7 as the coefficient of friction at the soil concrete interface. MUAP-12002 Rev 1, which

utilized this new coefficient of friction 0.5 was submitted in January of '13.

And revised responses to these RAI questions were submitted in March of '13 and are being reviewed by the staff.

CHAIRMAN STETKAR: I have a few questions about the flooding analyses. First of all a general question, does MHI or the COL have responsibility for evaluating flooding from sources in the essential service water pipe tunnel and the essential service water pipe chases?

As I read through the DCD it's not clear who owns those things, because they seem to be part of the standard design, but there's a lot of statements saying it's basically the COL's responsibility to evaluate them.

MR. BIERY: So your question is --

CHAIRMAN STETKAR: My question is who --

MR. BIERY: -- the division of responsibility?

CHAIRMAN STETKAR: In particular for, I mean in general for those, but in particular, because we're talking about flooding, for flooding from those sources.

MR. ITO: Okay. I'm --

CHAIRMAN STETKAR: I didn't see any, they weren't addressed in the DCC as far as I could tell, but I was just curious. At one level I'm concerned that they don't fall in the proverbial crack that the DCD presumes that the COL applicant will evaluate them while the COL applicant says, well they were part of the certified design so they've been evaluated over there.

So that's just sort of a general question. If you have a quick answer that's fine.

MR. ITO: I'm Masashi Ito. The variation of the ESWPT and the ESWPC is in COL applicant.

CHAIRMAN STETKAR: ESWPT, okay.

MR. ITO: Yes.

CHAIRMAN STETKAR: That's pretty clear. How about the ESWPC, the pipe chases?

MR. ITO: ESWPC is also in the COL applicant.

CHAIRMAN STETKAR: That's also COL, okay. Thank you. Let me just make a note here because otherwise I will forget. I was pretty sure the PT was part of the, the PT is acronym for the pipe tunnel, was part of the COLA applicant because that's connecting the ultimate heat sync to the pipe chases.

I really wasn't sure about the pipe chases, so, and you're saying they're both COLA, okay. So I'll

weigh on those questions now. Here are some details, and I just wanted to understand.

If I look, and I'll give you the reference so you can look it up, in DCD Section 3.4.1.5.2.1, that in particular addresses flooding in the reactor building radiological controlled area at elevation minus 26-feet, four inches down in the basement.

It's noted that the break in the charging piping on the West side of the radiological controlled area is the limiting flood, especially for the West side of that area.

You divide it up East and West and look at the flooding sources in the East and West and on the West side the charging line is the limiting flood. There's a statement saying that the water volume in that event, for the charging line failure, is calculated on the assumption that the release of water from the charging piping continues without being limited by the pump capability until the letdown line is automatically isolated.

And I read that and I thought about the system design and it's not clear to me why isolation of the letdown line will stop the flood from the charging line, because it's my understanding of the plant design that on low, low level in the volume control tank, the

suction of the charging pumps is automatically transferred to the refueling water storage pit so that when letdown is isolated the charging pumps will pump down the volume control pump, low, low level will occur, you'll transfer suction to the RWSP and those pumps will keep pumping water out that broken line for a long time until somebody turns off the pumps.

So I was curious about the justification that the volume of the flood in that particular location was limited by automatic isolation of the letdown line. You may want to double check that, or perhaps my understanding of the system design is not correct.

But I did, the reference that I looked at is Section 9.3.4.2.1 of the Design Certification Document, is what told me that the suction was transferred to the RWSP. So that's one question that I had about that flooding area.

And I'll let you take notes if you want to take notes because I'm a slow writer, also. On the East side of the non-radiological controlled area in the reactor building that contains the pump rooms basically, the analysis is done, it calculates a water depth of, I've forgotten what it is, 0.4 feet roughly, about five or six inches on the floor from flooding events over on that side of the building.

And it's noted that the assumption is made that water will flow from that portion of the non-radiological controlled area into the power supply building because the doors between the power supply building and that part of the non-radiological controlled area are apparently not designed to be watertight doors.

And, therefore, the flooding scenario, as best as I can tell, because if I look at the power supply building, the flood water depth is exactly the same height.

So what you've done is you've taken a flooding event of a certain duration, a certain volume, and you've distributed that volume, water volume, over the entire floor area of the non-radiological controlled area of the reactor building plus the floor area in the power supply building, as best as I can tell.

And I was curious whether that actually, on one level it sounds like it's conservative because flooding in either place will result in water in both places. On the other hand it may be non-conservative if those doors actually allow water to accumulate in one area.

So I was curious whether the conclusions regarding flooding in either area, either the power

supply building side of the door or the non-radiological controlled area side of the door, if those conclusions would be affected if you assumed those doors were watertight.

In other words, dividing those volumes and looking at the flooding sources. I don't know, you know, whether that's the case. I didn't try to do any calculations, but that's a question that I had.

This is a case where from one perspective an assumption that might sound like it's conservative could lead to non-conservatism if indeed those doors are fairly effective in terms of blocking communication, unless they're actually designed, I mean if they're designed to have, you know, large gaps underneath to allow water to transmit, that's a different story.

MR. BIERY: So your concern is uneven flooding levels?

CHAIRMAN STETKAR: My concern is, you know, you say those doors are not designed to watertight, well, suppose they are pretty darn good, let's call them not designed to be absolutely watertight, but pretty darn good, would that allow a water depth to accumulate in either of those areas if they were analyzed as individual compartments looking at the flooding sources in each such that the equipment might be compromised.

And the reason that I'm concerned, I'm actually more concerned about the non-radiological controlled area because those spaces contain two divisions of the component cooling water system and I've forgotten what else is in there, but I know that at least the component cooling water pumps are in there.

So if you flood out one of those areas that communicate to one another you could take out two divisions of equipment. And then the question is, if that occurs, can you safely shutdown if you assume one of the other divisions is out of service for maintenance and all of that kind of stuff?

So I was actually a little more concerned about the reactor building side of that door.

MR. BIERY: Do you want to answer now or do you want to wait?

CHAIRMAN STETKAR: And again, you may want to look into it, you know, this, you need to look at flooding sources and volumes.

MR. BIERY: Okay. I think we'll need to follow up --

CHAIRMAN STETKAR: Yes.

MR. BIERY: -- with you on those questions.

CHAIRMAN STETKAR: Yes. These tend to be, you know, we have a long history of doing this. You get

the kind of question out on the table and a lot of them are enough detail that they can't be answered in quick fashion.

I finally got it, yes, those common areas include two trains of the emergency feed water pumps and two trains of the component cooling water pumps. So it's the A/B trains in one area and the C/D trains in the other area.

When I looked at the flooding analyses in Appendix 3K, and again, I focused a little bit on those basement areas in the reactor building only because they're one of the few locations where you do have two trains of safety-related equipment in a common flooding area that's not isolated within itself by watertight doors or particular flood barriers.

And it's not clear to me, actually it is fairly clear to me that they're not considered, a couple of the flooding sources that I could think about in those areas, one notable source, is the essential service water connections to the component cooling water heat exchangers.

That's a large, potentially large volume of water because it's essentially a very large basin supply so you're limited, the amount of water that you put into to a compartment is limited basically by the amount of

time that the pump remains running.

And it wasn't clear whether the analyses accounted for flooding from that source. I don't think they do. And another potential source is the suction piping for the emergency feed water pumps from the emergency feed water storage pits which are up in a high elevation, but they're also large volumes of water.

So that, for example, if you had a break or a leak in that piping at the suction line to the emergency feed water pump, it seems like you could drain the whole emergency feed water pit into that room and it's not clear that those are evaluated either.

And if there's a reason for that I'd like to know.

MR. BIERY: So flooding from essential service water to the component cooling water heat exchanger and then suction piping from the emergency feed water pump feeding from the feed water pit?

CHAIRMAN STETKAR: From the original feed water pit, you know, upstairs. Now I know, before you say well this all safety-related, seismically qualified piping, I know that.

If I look at actual flooding events that have occurred in real operating experience, in many cases the floods have been associated with not necessarily

spontaneous failures or even seismic failures of piping or equipment, they've been associated with human errors during maintenance and things like that.

People have indeed put water from point A to point B under very creative ways because they forgot to open a particular drain valve or they forgot to isolate something.

So when I think about these flooding analyses I don't necessarily restrict myself to thinking about, you know, spontaneous failures of piping or things like that, which is another reason why I'm asking about this.

Because you can do maintenance during power operation on one division of equipment, you could indeed during power operation to have people working on those component cooling water heat exchangers or the emergency feed water pumps during power.

MR. BIERY: I think then --

CHAIRMAN STETKAR: So that's a different question.

MR. BIERY: I think we need to get back to you at a later time with that.

CHAIRMAN STETKAR: Okay. And that's all I had I think at that level on the flooding analyses. Anybody else have any questions? Any of you can speak

up at any time you want to. Okay, I'm sorry, go on, Mark.

MEMBER BROWN: Well I did have one minor question, very minor question, because as I look at the resolution of the RA on the main control room doors, the flooding --

MR. BIERY: Yes.

MEMBER BROWN: -- that was on Page 55, and I was trying to figure out, you all said you all updated the figure between the last revision in Rev 4 and I guess that was difficult to understand.

Some steps were added in Figure 4 into the two entrances into the main control room, it showed the steps and a little black line was put across the steps almost as if you anticipated opening the door with the hallway flooded but you could walk up the steps and have a little walkover and then get into the main control room without having any water come in.

So I have no idea what that Figure represented. This is not complex, I was just trying to understand how you all resolved that issue of providing a faux flow flood barriers at the main control room vestibule because both of them had watertight doors, the last Revision 3 as well as Revision 4.

And the only change is a little bit of a black line going up the stairwell area, or in the walk-in

area and now there's steps shown with a black line across it.

So is that, I'm just trying to figure out the philosophy, I mean, obviously, if you open the door in the main control room while there's water out there it would run in so I can understand having a watertight door and a small barrier walled entrance vestibule where you walk up something so that you maintain it above some height.

And I just wanted to know the philosophy of how you incorporated that?

MR. ITO: This is Masashi Ito of MNES. I think you are talking about the Figure 3K-5 --

MEMBER BROWN: 3K-5, yes.

MR. ITO: Okay.

MEMBER BROWN: And I looked at Rev 3 and I looked at Rev 4 --

MR. ITO: Yes.

MEMBER BROWN: -- trying to look at what the differences meant.

MR. ITO: Yes. So that black line added and up the stair, is the boundary of the flooding analysis --

MEMBER BROWN: I didn't get that.

MR. ITO: All right. The black line --

MEMBER BROWN: Open the door --

MR. ITO: Yes.

MEMBER BROWN: There's a watertight door off the hallway --

MR. ITO: Yes.

MEMBER BROWN: -- and now you show steps --

MR. ITO: Yes.

MEMBER BROWN: -- and then a little barrier, a little black line, goes across the stairwell.

MR. ITO: Yes. So in that 3.4 flooding analysis, that black line is the boundary of the --

MEMBER BROWN: What do you do step up over that or something? It's not another door. Is it a raised level?

MR. ITO: It's --

MEMBER BROWN: It's above the floor, it's above the hallway level enough to cover your flooding analysis?

MR. ITO: Correct.

MEMBER BROWN: Okay. If somebody opens the door, all right. Just trying to understand the schematic depiction.

CHAIRMAN STETKAR: I did have one more. I tend to look through my notes, given enough time.

MEMBER BROWN: I'll never do this again.

CHAIRMAN STETKAR: Hmm?

(Crosstalk)

CHAIRMAN STETKAR: No, thank you. Sometimes I need some delays. In the main steam feed water piping area in the reactor building at, I guess it's elevation 76, five inches, but anyway, it's the main steam feed water piping area.

The statement is made that there are doorways from that area to the other part of the non-radiological controlled area and it says that the bottom of the doorways is at elevation 76, five inches, which is 11-feet, five inches above the floor elevation of 65 feet, zero inches, and the conclusion is the doorways are located at a level that's higher than the flooding level estimated in those areas and, therefore, flood waters will not propagate into the other part of the reactor building.

The analyzed flood in that area is caused by a nominal one square foot break in the main feed water piping that accounts for eventual isolation of the main feed water line from whatever automatic signals will occur.

And I don't know enough about the configuration of the plant, so my first question is, is the main steam and feed water piping area open to the

turbine building or is it sealed from the turbine building?

In other words is it just an open, you know, pipe chase or pipe tunnel that if I'm standing in the turbine building I can look into it or is it actually sealed?

MR. ITO: It's not, no.

CHAIRMAN STETKAR: It's not which?

MR. BIERY: It's not because it's sealed or is it not sealed?

MR. ITO: Sealed.

MR. BIERY: It is sealed.

CHAIRMAN STETKAR: Okay. Now the question is then, if indeed this flood scenario occurs from a one square foot break in the main feed water piping, that break releases both mass and energy into this now what I understand to be an enclosed space.

And I understand if all of that, what starts out being some sort of two-phase mixture of steam and water condenses, you can calculate an eventual level of water in that area.

However, before that happens it seems to me that that area will experience a fairly reasonable pressure transient. I don't know what that pressure transient is because I can't boil water with help from

a stove, but the question that I had is if indeed this flooding event causes a pressure transient, are those doors designed to retain the pressure transient?

In other words, if they're not that break event will blow some mass and inventory out into the rest of the reactor building. So my real question here is not the elevation of the door for some sub-cooled level of water in the area, it's will the doors actually withstand the break?

I have a suspicion, but I'd like confirmation on that.

MR. BIERY: Do you want to answer now or?

MR. ITO: Can I clarify that? You are concerned that pressure onto that door?

MR. BIERY: Will the doors withstand the pressure loading?

MR. ITO: In the closed space?

MR. BIERY: Yes.

(Off the record comments)

MR. ITO: Yes, this also. I've got to check on that and get back later.

CHAIRMAN STETKAR: Get back to us?

MR. BIERY: We'll need to follow up with you on that one.

CHAIRMAN STETKAR: Okay. Thank you.

MR. BIERY: Okay, the next section we'll cover is Section 3.6, Protection Against Dynamic Effects Associated with Postulated Rupture of Piping. This DCD Section provides outline information on the pipe rupture protection.

The following are defined for the US-APWR, this includes design measures and bases, postulated pipe rupture location, jet impingement load, as well as leak-before-break evaluation methodology. There are no open items for this DCD Section.

In accordance with SRP 3.6.1, the plant is designed to provide protection against piping failure to ensure that these failures would not compromise the functional capability of safety-related systems nor maintain the safety of the plant when a pipe break is postulated, required plant conditions, design measures, and bases are provided.

Break or crack location is postulated in accordance with SRP Branch Technical Position 3-4, which is a highest stress location, location, great effect on essential equipment, et cetera.

Jet thrust reaction force and jet impingement loads are defined in accordance with SRP 3.6.2. Design criteria for pipe whip restraint and jet impingement barriers are provided.

Leak-before-break evaluation procedures, this subsection describes the design basis to eliminate the dynamic effects of pipe rupture for high energy piping systems of reactor coolant loop piping, RCL branch piping, and main steam piping.

The LBB evaluation is performed in accordance with SRP 3.6.3. LBB criteria are applied to RCL piping, branch piping, with normal diameter of six inches or larger as well as main steam piping inside a containment.

MEMBER RICCARDELLA: So do I understand that you've applied these before break to essentially all reactor coolant loop piping instead of pipe whip restraints? Instead of protection against pipe breaks?

MR. TAKAYAMA: We consider a percent, we apply LBB --

MEMBER RICCARDELLA: All locations? All locations in the reactor coolant system?

MR. TAKAYAMA: Oh, no. Greater than a 6-inch --

MEMBER RICCARDELLA: Yes. I understand. All locations greater than six inches.

MR. TAKAYAMA: Yes. But if you got a, is not satisfied even if the pipe size is 8-inch, all locations 8-inch pipe is post break is postulated.

MEMBER RICCARDELLA: Postulated, I understand. And are all these analyses complete? All the LBB analyses?

MR. TAKAYAMA: Every analysis completed.

MEMBER RICCARDELLA: Well then are there some locations where LBB was not satisfied?

MR. TAKAYAMA: Eight inch and/or 6-inch pipe diameter is not satisfied.

MEMBER RICCARDELLA: Yes, okay. I understand. Thank you.

MR. BIERY: Again, there are no open items for this DCD Section.

CHAIRMAN STETKAR: You essentially, in addition to the LBB inside the containment, you also take credit for no circumferential piping breaks in the main feed water and steam piping area, right?

MR. TAKAYAMA: Main feed --

CHAIRMAN STETKAR: Except for the branch lines, I mean --

MR. TAKAYAMA: But main feed water pipe we doesn't apply LBB.

CHAIRMAN STETKAR: Okay. Not in the PCCV, but out in the steam and feed water piping area, the largest break that you assume out there is that one square foot break.

MR. TAKAYAMA: Oh. I'll recap here --

CHAIRMAN STETKAR: Yes. I'm talking about the break exclusionary.

MR. TAKAYAMA: Yes. Outside of the --

CHAIRMAN STETKAR: Outside the PCCV, between the turbine building, the area that I was talking about and the flooding before.

MR. TAKAYAMA: Okay.

CHAIRMAN STETKAR: In that area you account only for circumferential breaks of the steam riser piping to the relief valves or the safety valves and a nominal one square foot break in the --

MR. TAKAYAMA: Yes.

CHAIRMAN STETKAR: The limiting one there is the feed water break, right?

MR. TAKAYAMA: Oh, yes. We don't operate LBB, that's the --

CHAIRMAN STETKAR: Right, but it's not LBB because you're not monitoring leakage there. This is just based on piping design, right?

MR. TAKAYAMA: Oh, yes, sir. Normally PCCV penetration breakage exclusion area. We apply PCCV containment break exclusion criteria we apply those portions main steam room.

CHAIRMAN STETKAR: Okay.

MR. TAKAYAMA: And one square feet break is assumed to break.

CHAIRMAN STETKAR: I understand that, and the one square foot break --

MR. TAKAYAMA: Yes.

CHAIRMAN STETKAR: -- that break area bounds the size of any penetration, any, what I call riser lines or drain lines from the steam and feed water piping, I at least looked for those.

The questions is, and I'm not a structural or piping systems engineer, I don't know the basis for declaring that area a break exclusion area. It's not the LBB criteria because, you know, you don't have the monitoring for leakage that you do on the other side of the wall to satisfy those criteria, this is just simply seems to be based on the design of the piping and the supports and structures in that area, is that correct?

MR. TAKAYAMA: Correct.

CHAIRMAN STETKAR: Okay. I'll ask the staff more about that later because, as I said, I don't understand anything about piping design. If there aren't any other members questions, I guess we can proceed, move on.

MR. BIERY: We'll now move onto Section 3.9, we're going to break this up into subsections.

We'll start with Subsection 3.9.1, Special Topics for Mechanical Components.

This subsection provides information on methods of analysis for ASME Code, Section III, Division 1, Classes I, II, and III components and supports to include core support structures. There are no open items for this subsection.

Design transients, components are evaluated using the design transients in accordance with the requirements for a Class I in ASME Code, Section III. The design transients give fluid system pressure, temperature, flow transients, and frequency to perform the ASME Code fatigue analysis and stress analysis.

These do not cover the seismic loading and other mechanical loadings on each component. Design transient item of each service level, A, B, C, and D, and test conditions addressed, 60-year design life is considered when determining the number of occurrences.

Computer programs, a number of computer programs, including commercial and in-house MHI codes, are used for static dynamic and hydraulic transient analysis for the component design. All computer programs are verified and validated in accordance with ASME NQA-1, Subpart 2.7.

NRC audits for computer programs was

completed back in August of 2011 and was subsequently closed. Again, no open items for this subsection.

CHAIRMAN STETKAR: Mark, this is not, I have a couple of questions. As I went through the Tables that list the number of transients that you analyzed, and I will fully admit this is not a question that really has anything to do with the safety evaluation of the equipment, it's more of a margins question.

And it really has more implication in terms of potential challenges to the operating utility of the plant than it does for any of the analysis work, so take it within that grain.

I'm not challenging the analyses, but if I look at a couple of the transients, in particular let me just focus on one of them, the other one has a little more margin.

If I look at the upset conditions, reactor trip from full power, there are three different categories of trips that are combined in that general classification. One is a reactor trip with no inadvertent cool down, which I'll call sort of a routine reactor trip.

One is a reactor trip with a cool down, but without a full safety injection and that's modeled by a steam generator overfill of that basically. And then

there's a reactor trip with a cool down and a subsequent safety injection which is modeled by I think a stuck open steam relief valve or something like that.

And the numbers of those events over the 60-year design life are 60 of the, what I'll call routine reactor trips, 30 of the cool down without a safety injection, and ten of the cool down with a safety injection.

And I kind of focused on the 60 normal reactor trips. That's basically one reactor trip per year which is for many plants a fairly large number, but not very large.

In other words, if I look at that margin of one trip per year for that particular transient compared to the margins that are built into your transient analyses for other things like ten to 100 events per year when I would expect to see maybe one.

I was curious why that one in particular had such a small margin? Because if I'm now a plant that's operating that has not the best experience in the world, and it doesn't have to get too bad to average one trip per year, I might be in trouble in terms of counting up transients later on and verifying that I need to redo analyses for this particular type of an event.

MR. BIERY: So you just want to know the

basis behind the number?

CHAIRMAN STETKAR: Yes. And, you know, to give people more margin would it make much difference to your transient analyses if you assume three, or four, or five of those events per year, which would be sort of more consistent with the margins that you have built into the other transient analyses that you've done.

I have no idea whether that trip is one of the more limiting transients. I just don't know. Pete?

MEMBER RICCARDELLA: Yes. A related question is, is it planned to install online fatigue monitoring at the plant?

MR. NISHIO: We are --

MALE PARTICIPANT: Might be a COL question.

(Crosstalk)

MR. TAKAYAMA: Did you say about fatigue monitoring system?

MEMBER RICCARDELLA: Yes.

MR. TAKAYAMA: Oh. Recently in the United States fatigue monitoring system is applied when planned, but it's the scope of past customers, so we may not have determined whether --

MEMBER RICCARDELLA: Okay.

MR. TAKAYAMA: -- in fatigue monitoring system is applied or it is not, but recently we believe

that customer will apply fatigue monitoring system.

MEMBER RICCARDELLA: This relates to Dr. Stetkar's question because what we've found in going back and looking at operating plants is that you have a relatively small number of some of these cycles, assume like 60 for the, but that the severity, when you look at how the plants operate they have many more cycles, but they're less severe.

And so rather than they just count the 60, most of the U.S. plants have installed fatigue monitoring systems that look at the actual severity of the transient.

So you might have, and that's acceptable in lieu of just counting your number of trips up to 60.

CHAIRMAN STETKAR: But as I understand your answer is, the decision to implement a fatigue monitoring program is basically a COL, that the certified design doesn't include credit for that if you will, is that correct?

MR. TAKAYAMA: So I don't know.

CHAIRMAN STETKAR: You did, your transient analyses do look at, they do look at some numbers of those more minor transients.

MEMBER RICCARDELLA: Yes.

CHAIRMAN STETKAR: They look at load

following type transients and things like that, so their analyses do indeed acknowledge that some of the less severe transients will occur at substantially higher frequencies.

MEMBER RICCARDELLA: Yes.

CHAIRMAN STETKAR: I mean a couple of times a day kind of frequencies. And I looked at those and those seem to be, you know, sort of reasonable margin at least for the design type calculations.

But there was, this one in particular is one that I said well, of all the ones that I thought about, this one seemed to have a relatively small margin compared to, you know, actual operating experience, which I grant you today is much, much better than it used to be, but plants still do have inadvertent, unexpected trips.

MR. BIERY: So normal reactor trips, you basically want to know the reasoning of the small margin?

CHAIRMAN STETKAR: Yes.

MR. BIERY: Are we prepared --

CHAIRMAN STETKAR: And as I said, it's not, your analysis is based on that number, so I'm not necessarily, I'm not questioning your analysis, I'm not questioning the safety of the plant, I'm not questioning the design of the plant even.

It's more of a question in terms of the amount of margin that's built into your analysis and the implications going forward for the eventual operator of that plant when they do start counting up these transients.

Because in most of the, in fact in all of the other cases, at least qualitatively, I can see there was a lot of margin built in there.

MR. BIERY: I think Hiroki Nishio wants to say something.

MR. NISHIO: The trip is for one unit. So the 60 is, I think 60 is an extensive trip, it's very, very conservative to the, and also we have the, looks at all the upright one unit --

CHAIRMAN STETKAR: Yes, I understand that. If I'm operating a reactor unit, this says on average you're accounting for one unplanned reactor trip without any further complications per year --

MR. NISHIO: Per year, yes.

CHAIRMAN STETKAR: -- over the 60-year life of the plan. And I'm saying that that is, for many currently operating plants, a larger number than they currently average, but it's not very much larger, okay.

It's probably not a factor of two larger, for example.

MR. SPRENGEL: So what's the --

CHAIRMAN STETKAR: It's an observation, Ryan.

MR. SPRENGEL: Okay.

CHAIRMAN STETKAR: I again, I'm trying to, you know, I'm an individual person just reading these things giving you comments and observations and this one in particular was notable because the amount of margin that's built into your analyses for this particular transient is much smaller than, much smaller than, the amount of margin built into your analyses for any of the other transients that you've evaluated, just looking at transient counts.

MR. SPRENGEL: Okay.

CHAIRMAN STETKAR: And you might of been out of the room, I prefaced it to say that I'm not raising an issue related to either of your analyses as being inadequate, your analyses are perfectly adequate.

I'm not raising an issue to say that you've obviously underestimated the number of transients based on any experience.

I'm just making the observation that this one in particular, the margin that's built into that, seems to be fairly small, such that going forward I could envision particular plant operators getting concerned

about having to redo the analysis because their projected number of reactor trips over the life of the plant will start to come very close to that number, especially if I think about early plant operations and projecting out.

So that's all. It was more of an observation.

MR. BIERY: We're going to switch seats here momentarily.

CHAIRMAN STETKAR: Be careful when you're moving up there, if you hit those microphones our recorder over there has very, very sensitive earphones and any time you hit this microphone it explodes in his ear.

So just be careful not to hit, they're really, really sensitive.

MR. BIERY: Yes, sir. I will now move --

CHAIRMAN STETKAR: By the time he turns 35 he will not have any hearing left, I'm convinced of this.

MR. BIERY: We'll go ahead and move onto Subsection 3.9.2, that's Dynamic Testing and Analysis of Systems Components and Equipment. The dynamic analysis methodology, a finite element model of the reactor coolant loop is used for analysis of RCL system response.

Additional finite element subsystem models for the steam generator and reactor pressure vessel are

applied for analysis of internal structures. Dynamic hydraulic loads during LOCA events are evaluated by a blowdown analysis using the MULTIFLEX Code.

Flow induced vibration assessment for reactor internals, FIV assessment program is established in accordance with Reg Guide 1.20. Conclusion of prediction analysis is that reactor internals have sufficient margins for adverse flow effects from fluid elastic instability, vortex-shedding lock-in, turbulence, and reactor coolant pump pulsation induced vibration.

Test program for the first operating plant, we'll use a vibration measurement and high-cycle fatigue evaluation conducted during the hot functional test.

There are a couple of open items for this subsection. First is to provide a comparison of technical data for both steam generators used in SONGS and US-APWR, explain any design differences between US-APWR steam generator design between US-APWR and Fort Calhoun, explain how contact force of the anti-vibration bars will be checked to ensure it is sufficiently high to prevent in-plane tube instability, as well as explain why the wear of the tube support plates is considered to be caused by turbulence excitation and not by in-plane tube instability.

This response is under preparation by MHI.

MEMBER RICCARDELLA: A general question related to this open item. Is it planned to do any testing of the steam generator tube bundles to verify adequate prediction against vibration?

MR. SPRENGEL: There is no testing planned right now.

MEMBER RICCARDELLA: Is the design significantly different than the replacement steam generators at San Onofre?

MR. SPRENGEL: I hesitate to answer that because that's a qualitative term --

MEMBER RICCARDELLA: Yes, I understand.

MR. SPRENGEL: -- but the response in the RAI actually focuses on differences between the different units --

MEMBER RICCARDELLA: Okay.

MR. SPRENGEL: -- and actually we're working with the staff to expand it beyond just on solely Fort Calhoun in terms of our comparisons. And I guess if there's any further detail or discussion that we want to have it would be a proprietary discussion.

MEMBER RICCARDELLA: Okay. We want to do that.

MR. SPRENGEL: We could discuss this

further, we just need to close the meeting.

CHAIRMAN STETKAR: Okay. I think, you know, if the members do have questions that they want to delve into, we have no problem closing the meeting at all.

So if we want to pursue this we can. I'll ask Pete and Sam, you said --

MEMBER ARMIJO: I have a question. I don't know how, you know, I only have one question. Well I hate to close the meeting for one question, but I think it's a good question, so it's up to you, John.

CHAIRMAN STETKAR: But you're pretty sure it would go into details of the design that are --

MEMBER ARMIJO: Well it relates, let me tell you what the question is and --

CHAIRMAN STETKAR: Yes, then we can decide.

MEMBER ARMIJO: -- then you can decide. But basically my question is, has Mitsubishi developed a test methodology, physical test methodology, that can reproduce the effects observed at San Onofre?

CHAIRMAN STETKAR: Sounds proprietary.

MEMBER ARMIJO: And has, you know, and that means have you actually done tests that said we can reproduce in our test facility the kind of wear that happened on those tubes?

And then the second part of that question

is, if you have such a test do you intend to apply it to the steam generators for this design?

MALE PARTICIPANT: What, a US-APWR --

CHAIRMAN STETKAR: Ryan already said that they don't plan to do any testing, right? So why don't, if there are questions, this is, we're probably treading into proprietary information is that right, Ryan?

(Crosstalk)

MR. SPRENGEL: We're also treading into --

CHAIRMAN STETKAR: Yes.

MEMBER ARMIJO: Yes, sensitive stuff.

MR. SPRENGEL: -- proprietary discussion that's not privy to MHI to disclose.

CHAIRMAN STETKAR: Yes. Okay. Let me ask MHI and the members, we have a window of opportunity. I can close the meeting, you know, now and we go till lunch to discuss steam generators if indeed that would be a productive use of 30 minutes of our time.

We could also close the meeting either today, this afternoon, or tomorrow to have that type of discussion, again, if it's productive. That's obviously, I don't whether it would be or not.

MR. SPRENGEL: Right. Well, unfortunately --

CHAIRMAN STETKAR: You basically have some

understanding of what you can discuss and, again, what's relevant to the US-APWR design.

MEMBER ARMIJO: Well, you know, I'd like the question to just stay on the record --

CHAIRMAN STETKAR: Yes.

MEMBER ARMIJO: -- and at some point we get it answered.

MR. SPRENGEL: Oh, no, I can answer that question. We have no testing planned for US-APWR.

MEMBER ARMIJO: Yes, but the prior question was, has Mitsubishi developed such a test or had someone develop it for them, a client, and demonstrate that they can reproduce what happened at San Onofre?

MR. SPRENGEL: I --

MEMBER ARMIJO: It's a very standard thing, you know --

MR. SPRENGEL: We can have that as a takeaway, but I think that question is actually for SONGS to answer.

CHAIRMAN STETKAR: That sounds more --

MR. SPRENGEL: I don't think I can answer that question.

MEMBER ARMIJO: Well it's a question for Mitsubishi, it's not for SONGS. I'm asking, do you have, it's very simple. I'll ask do you have, have you

developed a test that says hey, look, we had this problem with a steam generator, it wasn't detected or anticipated by analytical methods, but now we know what went wrong and we have developed a test to assure that we never get into that problem again.

That's really the question and it's not a SONGS issue, it's a Mitsubishi issue.

MEMBER RICCARDELLA: You know, and I could interpret that the ASME Code requires such a test. That if it's a nonstandard design and you're going to qualify your design, you need to do some testing.

MR. SPRENGEL: I don't feel comfortable continuing the discussion any further in an open session, so --

CHAIRMAN STETKAR: Certainly not in an open session. The question is can we continue it in a closed session where there could be some productive discussion?

If we can, we'll close it. Let's, look, rather than beating around the bush, let's close the session and discuss what we can discuss and at least air out more details of the questions and responses and then we'll at least understand each other a little bit better.

MR. SPRENGEL: Yes. I guess I would request that maybe we hold the closed portion at the, it might be difficult at the beginning of tomorrow. The

questions are trending away from US-APWR specifically, I understand the questions and I don't fault the questions, but it's a complex --

CHAIRMAN STETKAR: Sure.

MR. SPRENGEL: -- business environment right now that will be hard for us to --

CHAIRMAN STETKAR: You know, and I absolutely understand that, so, Ryan, I'll ask you, let me put on the table that we should close, we should have some closed discussion between now and the end of tomorrow to at least air out the questions and whatever we can discuss.

I'll let you decide on the timing of that and who you may or may not want present, okay?

MR. SPRENGEL: Okay.

CHAIRMAN STETKAR: So we'll just put that on the agenda --

MR. SPRENGEL: Sure.

CHAIRMAN STETKAR: -- and you can get back to us and tell us when you want that closed session and we'll make sure that it happens.

MEMBER SCHULTZ: John, I have a question for Mark that should be in open session on this topic. In this particular section as compared to the others that we've discussed this morning, you did not talk about code

qualification for the analysis that was performed or NRC audits of the analysis methodology as was done in other sections.

And I'm just wondering what the status is related to the qualification of the computational codes that have been used here and the NRC review.

MR. BIERY: Yes. I believe we covered, not in this section specifically, but in a prior section.

MEMBER SCHULTZ: Okay. So this applies across Section 3.9 then? That's this statement?

MR. BIERY: Yes.

MEMBER SCHULTZ: And into 3.9.2?

(Off the record comments)

MR. BIERY: I see agreement. It equalifies across the board.

MEMBER SCHULTZ: Okay, thank you.

MR. BIERY: The second open item in Subsection 3.9.2 is also steam generator related, provide preliminary design of the steam generator tube bundle and design of criteria for the tubes and retainer bars against flow-induced excitations, including various effects.

Again, response is under preparation. The next subsection is 3.9.3, that's ASME Code, Class I, II, and III components, components supports, and core

support structures. And we have two open items in this subsection.

This subsection provides regulatory compliance with the following, we list here 10 CFR 50, Appendix A, and 50.55(a), 10 CFR 50, Appendix S, and GDCs 2, 4, 14, and 15.

Component design, component design is performed in accordance with the ASME Boiler and Pressure Vessel Code, load combinations for all ASME service levels, such as dynamic loads and static loads are used.

Design criteria and stress limits are based on ASME Code Section 3. Reactor coolant loop design is presented in Appendix 3-C of Technical Report MUAP-09002, which is a summary of seismic and accident load conditions for primary components and piping.

Design specs for risk significant ASME Class I, II, and III PSCs, including supports, are provided in accordance with ASME Section 3. Design and installation of pressure relief devices, pressure relief valves are designed to comply with the requirements of ASME Code Section 3.

Pressurized or safety valves provide over-pressure protection for the reactor cooling system. Safety valves and power operated relief valves are provided on the steam lines.

Relief valves are provided on the residual heat removal system as well, and these perform low temperature over-pressure protection for the reactor cooling system.

Pump and valve operability assurance, pump operability, the operability of active pumps is established for safety-related functions and operating conditions. Active pump design criteria and stress limits for pressure retaining and support functions are based on ASME Code, Section III.

Similarly, for valve operability, operability of active valves is established for safety-related functions and operating conditions. And design criteria and stress limits for the pressure retaining support functions, again, based on ASME Code, Section III.

Component support design, is confirmed to be in accordance with the boiler and pressure vessel code. Load combinations for all ASME service levels, design criteria and stress limits are based on ASME Section 3.

Support classifications divide into two classifications, manufactured standard supports, to include spring hangers, snubbers, and struts, as well as supplementary steel supports, frame type pipe supports,

base plates and anchor bolts.

Again, we have some open items for this subsection. The first, requested MHI give details of the modeling and analysis methods of supports used for major components, including sketches of the support, design, loads, and load combinations, applicable stress of limited criteria, and fatigue evaluation criteria.

A response was submitted in April of 2009, and this RAI is an open item pending the staff's review of two Technical Reports, MUAP-08005 and MUAP-10006.

The next open item, requested MHI to inform the staff when PSC design specs within the audit scope are available for audit. MHI submitted revised design completion plan for the US-APWR piping systems and components in December of 2012, and its revision in August of '13, both of which requested the audit to be planned for December of this year.

MHI will submit a response to confirm completion of the design specs. Next subsection is 3.9.4, Control Rod Drive Systems. This subsection provides information of design, functional requirements and operability assurance program for the control rod drive mechanism portion of the control rod drive system.

The CRDM of the US-APWR is the magnetically operated jacking type based on the L-106A type CRDM,

which has been used in many operating plants in the U.S. and Japan.

The CRDM pressure housing is designed in accordance with ASME Code, Section III, Subsection NB. We have some open items for this subsection. The first requested MHI to show the basis of the allowable rod travel housing deflection and demonstrate that the estimated CRDM deflection in US-APWR does not exceed allowable limits.

In the amended response MHI estimated deflection of the CRDM pressure housing at Level D conditions, it's within the allowable limits. The response to these sub-questions is going to be revised to reflect updated seismic information at a later point. No impact on DCD is expected at this time.

Next open item, requested MHI to provide justification for the increase in the maximum CRDM deflection for the Level D conditions in the amended response to RAI 107-1293.

In this MHI explained that the CRDM dynamic response analysis results reflected the seismic conditions of DCD Rev 3. This question and response will be revised to reflect updated seismic information. Again, we also do not expect the DCD to be impacted by this.

Next subsection is Subsection 3.9.5, Reactor Pressure Vessel Internals. This subsection provides outline information on the structure and the design basis of upper and lower reactor pressure vessel internals.

The codes and standards applicable to reactor internals are ASME Code, Section III, Subsection NG, as well as ASME Code Section 11.

The US-APWR adopts the use of a neutron reflector replacing conventional baffle structures. This will result in improved neutron reflectivity and significant reduction in the number of threaded fasteners.

And we give a figure here of the neutron reflector assembly.

MEMBER RICCARDELLA: How do you reduce the number of fasteners? What is it about the design that reduces the number of fasteners?

MR. MATSUMOTO: Do you mean how many fasteners number? In the case of --

MEMBER RICCARDELLA: We can't hear you. You have to speak up.

CHAIRMAN STETKAR: Matsumoto-san, just speak up a little bit louder so that we can hear you on the recorder.

MR. MATSUMOTO: In the case of core barrier, fastener number is about 2000. On the other hand, this involve is about 20 piece.

MEMBER RICCARDELLA: Just 20 fasteners in the whole baffle structure?

MALE PARTICIPANT: Thank you.

MALE PARTICIPANT: As long as it hangs together. No RAIs on this one.

MALE PARTICIPANT: Okay.

MR. BIERY: Again, no open items for this subsection. Moving on to Subsection 3.9.6, Functional Design Qualification In-Service Testing Programs for Pumps, Valves, and dynamic Restraints.

ASME Code, Section III, Class I, II, and III safety-related pumps, valves, and dynamic restraints that are required to perform a safety function are subjected to in-service testing to assess and verify operational readiness as set forth in 10 CFR 50.55(a)(f), as well as the ASME OM Code.

The in-service testing program implements ASME OM Code 2004 edition through '06 addenda. The COL applicant is responsible for administrative control of the IST program as well as control of the ASME Code addition and addenda which will be used in their IST programs.

We have an open item for this subsection. The RAI is an open item to track and audit of the US-APWR design and procurement specs to evaluate implementation of the functional design and qualification and IST programs.

And currently preparation of sample design and procurement specs isn't progressed by MHI.

CHAIRMAN STETKAR: Mark, I, and I don't know the best way to do this. I went through Table 3.9-14. I really didn't intend to go through it, but I got started and one thing led to another, so I did.

And I had several questions about the testing frequencies for a number of valves listed in that Table that in general, you either go for quarterly testing or sometimes it's referred to as cold shutdown, sometimes it's referred to as a refueling outage, sometimes it's referred to, you know, 24 months, but it's basically 2-year testing or quarterly testing on the vast majority of those valves.

And in many cases the 2-year testing interval for valves was justified by statements saying either of the tests would result in disruption of a normally operating support system or it could result in a plant transient and therefore the test should only be conducted at cold shutdown or when those systems were not

normally pressurized or operating.

And I looked at a number of those, and I actually have a lot of questions here, which are probably too detailed to go through, but in many cases it wasn't clear to me that the justification was consistent with the way that the plant will be operated.

For example, you know, if I look at a lot of the support systems, essential service water, essential chilled water, component cooling water, ventilation systems, the plant is basically designed to have two trains of those things running normally, two trains in standby.

And there's a requirement for the pumps or the fans, or the chillers, or whatever the large rotating equipment, that that equipment be tested once per quarter, that's the Table 3, I've forgotten the Table number. It's the Table on the pumps and those sort of things.

I don't know how the COL applicant is actually going to run their plant. A lot of plants that have those quarterly testing requirements take the normally running train and put it in standby and take the standby train and make it normally running.

In other words, they switch the plant configuration once per quarter, that's good, they

satisfy their testing requirements and they balance the run times on pieces of equipment, so everybody's happy, but that's obviously the COL applicant and how they're going to run the plant.

But given the fact that half the plant is in standby at least half the time, it's not at all clear to me why I can't stroke a valve for something that's in standby because stroking that valve would disrupt a normally running cooling system.

And yet the justification is made that if I operate those valves I'll disrupt a normally running system, so, therefore, I can only do it during cold shutdown. And there are a lot of other things like testing of the safety depressurization valves and depressurization valves from the pressurizer.

It says that I have to test the block valve once per quarter, but I can't test the other ones unless I'm in cold shutdown because, well I might induce a LOCA. You'd only induce a LOCA if the block valve's closed.

Close the block valve, cycle a depressurization valve, re-open the block valve. You have to test the block valve. So I had a lot of questions like that that I couldn't understand the justification for not testing once per quarter on a lot of the systems.

As I said there's a lot of details, and this

really, I'll telegraph it to the staff because I didn't see any staff questions regarding those test frequencies.

So I'll just raise it now, if you have some general response, you know, rather than going into valve-by-valve, system-by-system, I'd appreciate it, from your perspective, and I will ask the staff, you know, sort of a similar question when they come up about why they didn't have questions about those test frequencies in the Table.

MR. BIERY: So your concern is consistency and reasoning behind the testing frequencies of those valves?

CHAIRMAN STETKAR: Exactly. And in particular wherever the testing is justified by statements that says performing the testing once per quarter would disrupt a, let's say, normally required support system.

And it's typically associated with cooling water or ventilation chilled water systems, those types of justifications, or that performance of the test would result in a plant transient and those justifications are typically assigned to things like the safety depressurization valves, the depressurization valves on the pressurizer, and the main steam relief valves, the

main steam depressurization valves over on the secondary side of the plant.

MR. BIERY: Okay. I think we'll need to look into that further. I don't know if we can answer that immediately.

CHAIRMAN STETKAR: If you want certain, you know, if you want specific examples for elaboration, I can give that you offline --

MR. BIERY: Okay.

CHAIRMAN STETKAR: -- but that detail tends to get way down in the noise level for discussion, you know, in the full open session here.

MEMBER SCHULTZ: Mark, just to ask you a question about this particular response that you have in preparation, in progress --

MR. BIERY: Yes.

MEMBER SCHULTZ: The NRC concern seems to be in the detail associated with evaluating the implementation and the way you've described it you're going to hit the upper level, first part of the concern, I just wanted to make sure that it's going to follow through to look at functional design qualification and also the IST programs.

This is one question for this section and so I'm assuming it's going to cover, your response is

going to cover each of those elements in some level of detail.

MR. BIERY: Yes. the sample design specifications that we're preparing for this audit are intended to address all these concerns making sure that we're consistent for a licensing basis in these specs, and including all these points in the evaluation.

MEMBER SCHULTZ: Good. Thank you.

CHAIRMAN STETKAR: I think since you have several slides that address the seismic qualification, I think what I'll do is we'll break for lunch now rather than trying to rush through those.

We have something else happening at noon, so we basically have to break now. So what I'll do is we'll recess for lunch and reconvene at 1:00 p.m., according to the schedule, and finish up this part of the presentation.

(Whereupon, the foregoing matter went off the record at 11:55 a.m. and went back on the record at 1:01 p.m.)

CHAIRMAN STETKAR: We are back in session and we'll continue with MHI's presentation on Section 3.10, I guess, although it looks like you may have some answers.

MR. BIERY: Yes, sir.

CHAIRMAN STETKAR: So I'll keep my mouth shut and let you speak.

MR. BIERY: Yes, sir. Good afternoon, gentlemen. My name again is Mark Biery. Before we proceed to Section 3.10 we're going to circle back and answer some of the questions from 3.5 and I'll go ahead and I'll turn it over.

MR. MINAMI: Yes. My name is Minami, again, and I understand your question is how I calculated the probability of that past due to low-cycle fatigue.

And the calculation procedure is explained in the Section 3.3 in the MUAP --

MALE PARTICIPANT: Yes.

MR. MINAMI: -- in 07028. And on this slide there is an equation which is usually called Paris equation and usually used for the low-cycle analysis fatigue.

CHAIRMAN STETKAR: Could you just make sure that you speak up so that we catch on the record. You can move the microphone a little bit closer. It's just really important that we pick up your words. Thank you.

MR. MINAMI: And then this equation "A" denotes crack size and the "N" is the number of where the start and stop cycle and indicates stress intensity. And the "C0" and the "N" is a constant particular to the

material, right?

MALE PARTICIPANT: Yes.

MR. MINAMI: And from this equation we can derive the following next equation and that's equal to blah, blah, blah. And based on this equation --

MEMBER RICCARDELLA: But in the second equation you need to establish a critical, a crit --

MR. MINAMI: Right.

MEMBER RICCARDELLA: -- which is a function of the fracture toughness, right?

MR. MINAMI: K1C, function of K1C, and as for the K1C we used that propriety number for estimating a critical and based on the, we used Belgrade method, which is authorized by --

MEMBER RICCARDELLA: Of the what method?

MR. MINAMI: Belgrade. And we obtained K1C using the minimum material property, which is specified in purchase specification.

MEMBER RICCARDELLA: Okay. So you didn't assume --

CHAIRMAN STETKAR: You two, pull the mike closer to you so you come across, I'm sorry.

MEMBER RICCARDELLA: I understand, keep going.

MR. MINAMI: So we estimated K1C based on

our purchase specification of the low pressure turbine rotor material. And based on the K1C we estimated a critical, a critical, right? So if we obtain a critical we can estimate the number of the start and stop to reach a critical from the initial size of the crack, right?

MEMBER RICCARDELLA: Yes.

MR. MINAMI: And for obtaining the probability of the rotor failure due to low-cycle fatigue we used in some variation for C0 and based on our own experience.

MEMBER RICCARDELLA: Okay.

MR. MINAMI: And the variation is shown in the Table 3.3, this one.

MEMBER RICCARDELLA: Yes.

MR. MINAMI: Right?

MEMBER RICCARDELLA: I saw that.

MR. MINAMI: Right. And using this mean value understand that deviation. We estimated the number. So if we use this variation the number of the start and the stop to reach the critical speed depend on the combination of C0 and the N, right?

MEMBER RICCARDELLA: Yes.

MR. MINAMI: And if we assume some profile of that C0 and zero we can get to the probability of occurrence.

MEMBER RICCARDELLA: Well you can get a distribution on N sub F on the number of cycles, right?

MR. MINAMI: Right.

MEMBER RICCARDELLA: But then how does that relate to the probability of a rupture?

MR. MINAMI: If we estimate one combination of C0 and the N we will obtain one COG, COG means variation between crack size and the number. And if we add in another combination of the C0 and the N we'll have another variation.

MEMBER RICCARDELLA: Yes.

MR. MINAMI: So, I mean, we have the probability of occurrence depending on the number.

MEMBER RICCARDELLA: I see. So you have all these curves and then you expect 60 cycles or some number of cycles --

MR. MINAMI: Right.

MEMBER RICCARDELLA: -- and you put that in there, I see.

MR. MINAMI: Right.

MEMBER RICCARDELLA: Okay.

MR. MINAMI: Look at the way we obtain the probability of most failures.

MEMBER RICCARDELLA: I understand, that's reasonable. I would only say that in my experience

there's also a probability of distribution on the fracture toughness --

MR. MINAMI: Right.

MEMBER RICCARDELLA: -- and also on the initial crack size because you have some uncertainty in the inspections that --

MR. MINAMI: Right. And for the initial crack size we used some numbers, but it is equal to the maximum undetectable size and we assumed some additional margin to compensate some uncertainty.

MEMBER RICCARDELLA: Okay. All right. Yes, so --

MR. MINAMI: So --

MEMBER RICCARDELLA: Okay. So you treat the uncertainty in C and the N and then as far as the uncertainty on initial crack size and on K1C you just assume conservative bounding values.

MR. MINAMI: Right.

MEMBER RICCARDELLA: I understand.

MR. MINAMI: Right.

MEMBER RICCARDELLA: Thank you.

CHAIRMAN STETKAR: Thank you. That was a very good explanation.

MR. MINAMI: Thank you.

CHAIRMAN STETKAR: That helps a lot. It's

another reason why we kind of schedule these meetings for two days because our experience has been that a lot of times we can get some of the questions answered pretty much in real time.

So that's useful, thank you. And I guess, Mark, you can now continue.

MR. BIERY: Yes, sir.

CHAIRMAN STETKAR: Do whatever you want to.

MR. BIERY: Go ahead and switch it back, please. Switch to the other computer, please.

CHAIRMAN STETKAR: By the way we'll need this slide for part of our record --

MR. BIERY: Yes.

CHAIRMAN STETKAR: -- because it's part of the record of the meeting, so make sure we have that.

MR. BIERY: We'll proceed as soon as we get our switched computer back.

CHAIRMAN STETKAR: Yes, that's fine.

(Pause)

CHAIRMAN STETKAR: Excellent.

(Crosstalk)

MR. BIERY: Okay. We'll go ahead and proceed onto Section 3.10 and that is Seismic and Dynamic Qualification of Mechanical and Electrical Equipment.

Qualification standards used in this

section include IEEE Standard 344-2004 as modified by Reg Guide 1.100 for safety-related mechanical and electrical equipment and supports.

Also used ASME Code, Section III for structural integrity of safety-related pressure boundary components. And then finally, ASME QME-1-2007 for qualification of active mechanical equipment.

Performance requirements for Seismic Cat I instrumentation electrical equipment are provided in corresponding equipment qualification summary data sheets, or EQSDSs.

Seismic Cat I active mechanical components are defined in corresponding equipment specs along with system functional requirements, performance criteria to perform their designated safety-related functions under the postulated SSC in combination of other concurrent loadings, deformation of supports and structures is considered acceptable, provide their's and other equipment safety-related functional performance are not compromised.

Testing is in accordance with Reg Guide 1.100, Rev 3. The EQ Program is provided in DCD Section 3.11. Analyses are done in accordance with Reg Guide 1.100, Rev 3. Analysis without testing is acceptable only if structural integrity alone can ensure the design

intended design function.

Combination of testing analysis, again, in accordance with Reg Guide 1.100, if it's utilized when the equipment cannot be practically qualified by analysis or testing alone.

Test or analysis to assure structural capability, including anchorage, electrical equipment instrumentation supports are tested with the equipment installed or an equivalent dummy. For mechanical equipment supports, including pumps, valves, valve operators, and fans, it's in accordance with ASME Code, Section III.

For instrumentation line supports using the criteria from ASME Code, Section III, Subsection NF, for equipment Class I and II supports. Equipment qualification file will include qualification method used for equipment, tested analyses results, lists of systems equipment, equipment support structures, EQSDSs, seismic input requirements, experience-based qualification will not be used for any equipment.

We do have some open items for this DCD Section. The first, requested MHI to provide a list of components in the gas turbine generator system to be seismically qualified with the method of seismic qualifications specified for each component in estimated

qualification schedules.

The staff had an opportunity to witness the testing. And then also, in DCD Tier 2, Section 3.10, the seismic qualification criteria and procedures including referenced report number, formulated electrical and mechanical components of the gas turbine generator.

And then MHI submitted a revised response in June of '13, and this RAI remains an open item pending submittal and the staff review of MUAP-10023 revisions, Revision 6, submitted in September '13, and we have another revision scheduled for December of '13.

MEMBER RICCARDELLA: So to go back to an earlier discussion, you asked well this turbine generator, the gas turbine generator is not seismic class, but yet you are going to qualify it. Is that --

CHAIRMAN STETKAR: There's two sets of gas turbine generators.

MEMBER RICCARDELLA: Ah.

CHAIRMAN STETKAR: These are the emergency gas turbine generators.

MEMBER RICCARDELLA: Okay.

CHAIRMAN STETKAR: The one's I was talking about is a different set, they're the alternate AC gas turbine generators.

MEMBER RICCARDELLA: Backup system or

backup --

CHAIRMAN STETKAR: They're --

MEMBER RICCARDELLA: Okay.

CHAIRMAN STETKAR: -- non-safety-related backup for a station blackout.

MEMBER RICCARDELLA: Okay.

MR. BIERY: With that we'll move onto Section 3.12, piping design review, and we're going to go ahead and switch seats here quickly again. This section covers the design of the piping systems including piping supports which comprise Seismic Cat I and Non-Seismic Cat I piping systems.

Codes and standards are consistent with 10 CFR 50, Appendix A, General Design Criteria 1, 2, 4, 14, and 15, as well as 10 CFR 50, Appendix S. Piping analysis used the 1992 edition with '92 addenda of the ASME Code, Section III, Division I, that's Subsections NB, NC, and ND, in accordance with the requirements of 10 CFR 50.55(a).

Material properties comply with ASME Code, Section II, 2001 edition through, including, 2003 addenda.

MEMBER RICCARDELLA: Is that the most recently approved version of Section III? It seems old.

MR. TAKAYAMA: 1992 edition is because 10

CFR 50.55(a) doesn't induce 1993 edition. So we use 1992 edition.

MEMBER RICCARDELLA: Okay.

MR. BIERY: Analysis methods, seismic analysis for all Seismic Category I and II piping systems use methods in accordance with SRP 3.7.3. These methods include the response spectrum method or, where applicable, the equivalent static load method, or modeling supports in the piping analysis, the decoupled support model, model the stiffness, or an integrated support model, the actual structural model of the support is used.

Dynamic analysis consists of a sequence of nodes connected by straight pipe elements. Curved pipe elements with stiffness properties representing the piping, as well as other in-line components.

Computer programs are verified and validated using NUREG/CR-1677, Volumes I and II. NRC audit for the computer program and design methodology is documented in DCD Section 3.12. That audit was completed in August of 2011.

Stress analysis criteria, allowable stress is a piping system for various loads and load combinations are defined based on ASME Code, Section III, Subsections NB, NC, and ND.

Combination of modal response, SRSS 10 percent grouping method in high frequency modes are considered in accordance with Reg Guide 1.92. Class I piping is evaluated for the effects of fatigue caused by thermal transients and other cyclic events, including earthquakes and thermal stratification occurring in the surge line.

Thermal stratification and oscillation in the closed branch piping connected to the RCS would be prevented in accordance with US-APWR design approach described in DCD Subsection 3.12.5.9, and that, again, is DCD Revision 4.

The environmental impact on fatigue of Class I piping follows the requirements delineated in Reg Guide 1.207. Seismic Category I pipe supports are designed in accordance with Subsection NF of the ASME Code, Section III, 2001 edition through, including, 2003 addenda.

The load combinations for the piping support design are defined based on Level A, B, C, and D service conditions. And we have some open items for this DCD Section.

The first, requests MHI to clarify the analysis model and input for reactor coolant loop piping dynamic analysis. A response was submitted with the

requested clarification in September of '12. The seismic and accident design analysis method for RCL system components is presented in MUAP-09002, and the next revision will be submitted this month.

There are DCD changes, standard response, which have been incorporated into DCD Rev 4 and that is still pending staff review and, therefore, this RAI remains an open item pending staff review.

The final section for today will be Section 3.13, threaded fasteners. The design requirements are the same as that for existing operating plants. Material selection is based on ASME Boiler and Pressure Vessel Code, selection criteria are in accordance with NCA-1220 and NB/NC/ND-2128.

Fracture toughness is in accordance with NB/NC/ND-2300 and 10 CFR 50, Appendix G. Lubricants include Fel-Pro, Neolube #126, and Nuclear Grade Neverseez. MoS2 and copper-based anti-seize compounds will not be used.

And then reactor vessel closure stud bold requirements, again, similar as to the existing plants. In-service inspection requirement, that procedure is based on ASME Code, Section XI, and then there are no open items for this DCD Section.

And then that concludes our presentation

for today.

CHAIRMAN STETKAR: Okay. We're well ahead of schedule, sometimes that happens. Do any of the members have any questions for MHI on any of the subsections of Chapter 3?

Nothing else? If not, then --

MEMBER RICCARDELLA: Just one general question, and I noticed, and this might be part of what's going to be reviewed in 3.7, 3.8, but I noticed in the load definition document that your seismic peak ground acceleration was 0.3g. Is that OBE or SSE, 0.3, service Level B or service Level D?

MR. TAKAYAMA: SSE.

MEMBER RICCARDELLA: SSE, okay. 0.3g is SSE. Thank you.

CHAIRMAN STETKAR: You selected OBE as --

MR. TAKAYAMA: One-third.

CHAIRMAN STETKAR: -- yes, one-third of that or something like that.

MR. TAKAYAMA: Well, okay.

MEMBER RICCARDELLA: One-third?

MR. TAKAYAMA: Little less than one-third ACC -- OBEs.

CHAIRMAN STETKAR: Okay. Any other questions? If not, thank you for covering an awful lot

of material very well. I really appreciate it. We obviously had a few questions that we'll need to follow up on.

The staff is up next and I don't know whether you have people here yet because we're about an hour and a half ahead of our published schedules and people tend to organize their day around the meeting agenda.

So, Dennis, what's your situation?

MR. GALVIN: I told people to be here as early as 2:00 p.m., but some people may be waiting in their offices, we could --

CHAIRMAN STETKAR: I tell you what, why don't we, so we're just not sitting around on the record, why don't we take a break and see when you can muster folks here and then I'll ask people -- let's take a break until 1:45 p.m. and hopefully you can get a few folks to start your presentation by then.

Does that seem feasible?

MR. GALVIN: I think that's feasible.

CHAIRMAN STETKAR: Okay.

MR. GALVIN: Okay.

CHAIRMAN STETKAR: Good. Let's do that. Let's recess until 1:45 p.m.

(Whereupon, the foregoing matter went off the record at 1:26 p.m. and went back on the record at

1:46 p.m.)

CHAIRMAN STETKAR: We are back in session and we'll hear from the staff on their review of DCD Chapter 3. Dennis, it's yours.

MR. GALVIN: Hello, my name is Dennis Galvin. I'm the Chapter 3 Project Manager. As you'll see, because we -- here's the staff. We have two slides with the staff, who contributed to the review. As you can see, we have 24 open items. About seven different, over seven different technical areas.

So what we're going to do, is I'll go ahead and introduce the open items, give a brief description. If you want to go ahead and if you have any questions, we'll have the technical staff, go ahead and answer the questions. Just so they don't have people pairing, they'll probably go to the mikes to answer the questions.

So first open item is, this covers three different Sections, 3.2.1, 3.2.2, and 3.9.3. We need, the staff does an audit of the design specifications and also as part of the review of, ASME Class 1, 2, and 3 components. MHI as they told you, has been preparing this design specifications.

We also are going to combine that with a performance specification audit. Those were scheduled to be completed, available in January 2014. At this

point, with the slowdown, this is not currently scheduled. We learned that in the last week.

The next two we've already covered, are flooding, RAI 841-6055. This RAI tracks the need for the staff to complete a flooding analysis audit concerning the modeling assumptions and pipe break selections. Staff has completed an audit, and MHI has submitted a follow up response. And we're still reviewing the response.

Also again the staff identified, next one, question 30, the staff is awaiting the submittal of changes to the building layout, flood barriers, and water-tight doors resulting from seismic design changes. Those changes have been submitted and the staff is in the process of reviewing those.

CHAIRMAN STETKAR: Dennis, while you're talking about flooding, I have a question. I'm trying to find my notes here. There, in MHI's discussion of flooding from internal sources, they invoke the notion that cracking in certain moderate energy piping need not be assumed. In particular, I'll give you a quote out of Section 3.4.1.3 of the DCD. It says, "most of the water containing moderate energy piping in the radiological controlled area of the reactor building, is excluded from flooding source because that piping is to be designed so

that crack is not required to be postulated in the line, in accordance with the criteria described in Subsection 3.6.2.1.2.2. This is attained by maintaining stress on the pipes below the threshold by means of route and support design."

And it's my understanding that they basically used that assumption to exclude a large numbers of potential flooding sources. I asked the applicant this morning about how they consider potential flooding, for example from the emergency service water, or the essential service water piping connections to the component cooling water heat exchanges? It's clear that they don't.

I asked them how they consider potential flooding from the connections from the emergency storage tanks to the emergency feed water pump suctions? It's clear that they don't. If they did, they would have flood depths higher than they do.

So my question is, given the fact that piping can crack and more importantly, my observation earlier this morning, that operating experience has shown that floods often result not from spontaneous cracking of pipe, but from maintenance and operational activities that drain the water from a point A to a point B. How's the staff, the staff has apparently accepted

this notion of none fallible piping as a basis for the flooding analysis, I was curious about how that works out?

MR. GALVIN: We have Angelo Stubbs from the staff here that --

MR. STUBBS: Okay. They came, this is something that they came in and they made some changes to what they were looking at with the moderate energy line cracks, from the original application. Where they were going to do a redesign and I'm not, to say is, there are certain sections and in some cases they may be complete systems where they actually looked at that. So that they would do the design, so that they wouldn't see certain stress levels.

I want to, and based on that, there are certain areas where they re-evaluated the results of flooding due to moderate energy line cracks. The acceptance on that, I mean, I going to let Renee talk to that from the ME's stand point. But there was a criteria at 3.6.2 that decided that if they need that criteria, they would not necessarily have to assume a leak in those sections of the pipe. Renee can talk about it.

MR. GALVIN: This is Renee Li.

MS. LI: Yes, I'm Renee Li from Mechanical Engineering Branch. I think this also from this morning

discussion, we also have some question related to break exclusion. So because of related, I will try to address both issues. So first, the staff guideline that, for postulating either breaks, in high energy line, or cracks in moderate energy line is provided in SRP 3.6.2. And in associated Branch Technical Position 3-4.

And SRP 3.6.2 and Branch Technical Position 3-4 in part, it provides guideline of where to postulate breaks and cracks in high energy line, as well as moderate energy line. And approach in those guideline is a mechanistic, we assume the mechanistic value which means, the failure not induced by any other event or source, like seismic.

So it's a deterministic criteria and in final, the ASME quote stress and fatigue design criteria. So the Branch Technical Position 3-4 provide guideline of a stress threshold. If the piping design maintain the stress and the fatigue usage factor, without those threshold, then one does not have to postulate breaks in high energy line and cracks in moderate energy lines.

But I want to stress that, those breaks and cracks that covered within the scope of 3.6.2, like I say, is a mechanistic value. It does not include failure due to seismic or other, you know, like rusting, corrosion, erosion. It does not consider those.

Now as far as the question you just asked about moderate energy line, within the scope of 3.6.2 and the Branch Technical Position 3-4, it then depend the stress and fatigue within that threshold for the moderate energy line, then they don't have to postulate leakage, crack or assume a crack not would result in flooding.

But okay, flooding can be induced by other, for example if you have a non-seismic category piping, even it's another high energy line, since it's not designed to seismic event, therefore one has to consider flooding that pipe breaks, of those moderate energy lines during the seismic event. So I want to differentiate the importance of this too. And since 3.6.2 going to come up, as far as the breaks exclusion --

CHAIRMAN STETKAR: Let me, let's, if I can let's address that because it is a bit different in my mind than the concern I've raised. I hear what you're saying, and if I'm a piping designer, I understand what you're saying about pipes not cracking by themselves.

My concern is broader than that. I'm not, my concern is looking at a location in the plant where I can have two of my safety divisions of equipment, disabled, potentially, by a flood.

Because they are in a, as best as I can tell, shared space. Those are A and B, of the emergency feed

water, with A and B of component cooling water. Both of those being safety-related systems.

And the C and D divisions being in another shared space. So if I can flood the shared space occupied by trains A and B, I take out one half of my plants safety-related equipment.

Now my technical specifications allow me to have one train out of service, indefinitely. So the question is if I have a flood that takes out half of my safety-related divisions with one of the other trains out of service, can I indeed achieve safe shutdown? That's a question in my mind.

Then I start to think about how I can get floods? Well if you say that none of the potential water sources have sufficient capacity to fill that room to a depth so that my equipment can be submerged, or affected, that's one type of analysis. I'm willing to accept that analysis.

I just don't have the inventory, no matter what I do to try to get the inventory to that location. I submit that there are a couple of sources for those locations, namely the emergency feed water storage tanks, storage pits. And the essential service water systems that can probably fill those rooms pretty deeply.

I haven't done the calculation myself,

because I don't have all the dimensions. But it's clear that they have not been considered as flooding sources. And my question is why collectively are we not evaluating those potential flooding sources?

I understand that we're not evaluating them from a piping design criteria, because we design and construct piping very well. I'm more concerned about everything that can happen at a power plant, and our experience that indeed we have flooded locations from inadvertent operations or maintenance activities.

And having confidence that this plant design is not vulnerable or unduly vulnerable to those types of events. And I don't know where that's covered. Because this is the only place that I can see where a flooding analysis is performed, other than the PRA.

MR. STUBBS: Okay, I think I understand --

CHAIRMAN STETKAR: Which is not really done a probabilistic flooding analysis, but that's a different issue.

MR. STUBBS: I think I understand what you're saying --

CHAIRMAN STETKAR: Okay.

MR. STUBBS: -- where you're coming from, but I guess we look at our guidance. It sort of goes beyond what we require to be postulated. And I guess

that's what we're focused on.

CHAIRMAN STETKAR: Yes, and I hear what you're saying. The introduction basically set that stage.

MR. STUBBS: But, in number of, you know, having conducted an audit, they do take into consideration, I know for a number of systems, we didn't look at every one we did, where there are backup water sources and things are switched. And in many cases they did take into consideration that if, one water source is depleted, or you can switch to another water source.

And they, I think it's in some cases they assumed a certain amount of time before operator action, being 45 minutes or whatever, before operator action would be used to cut off the secondary source.

But I can't go for every system, and go through as part of the audit. We did look at situations where just because you had, where something may be either automatically or manually transferred to a different source, water source, which could continue the flooding.

CHAIRMAN STETKAR: Let me --

MR. STUBBS: I don't know if, with the EFW pits, I think they're like 200,000 gallons of water available in those pits, and the lines were that, the whole idea is to have the design so that you don't, this

is when you're most, really --

CHAIRMAN STETKAR: Quite honestly, you know, as I kind of walked through those analyses and I looked for locations in the plant where you might be vulnerable to failing more than one division of equipment from flooding.

And in most parts of the plant, the flooding barriers, the water tight doors, and the flooding barriers are designed such that even if I filled a compartment full of water, you would at most disable a single division.

There are only a few locations that I could identify where there might be a vulnerability to flooding two divisions of equipment. The basement of the non-radiological controlled area in the reactor building is one. Then there's an upper elevation where you have some shared spaces for ventilation equipment.

MR. STUBBS: Right.

CHAIRMAN STETKAR: Were about the only two locations that I could equivocally come up with --

MR. STUBBS: Yes, I think those are the only two.

CHAIRMAN STETKAR: And the upper divisions may not, it, I didn't raise that because the upper divisions I couldn't necessarily figure out where the

water was going to, a large volume of water was going to come from. But down in the basement, because of the two water sources and the fact that it is the basement --

MR. STUBBS: There's no divisions in that place though.

CHAIRMAN STETKAR: I was, I am really curious about that. It may be an issue in terms of the guidance and that may be it. But I'll still raise it as a question at least in my mind about what is the vulnerability of flooding in those areas. Can we indeed disable two trains of equipment from a possible flooding source in those areas?

Irregardless of trying to postulate, is it going to be a pipe breaker, or is it going to be a crack, or is it going to be leaking flange or is it, you know, regardless of how I can get the water in there.

MR. STUBBS: I, in that case, well the only thing without the guidance, yes, but no, we were going to isolate it.

(Crosstalk)

MR. STUBBS: What margin do we have to, when something, you know, it's a --

CHAIRMAN STETKAR: Okay.

MR. STUBBS: -- that's all I'm saying I guess is to really look at. Do we have, you know, what

sort of margin do we have? Six inches of margin. But there would still need to be some intervention from someone to stop the flooding at some point, otherwise we're going, like I said, and that would disable equipment. When you have, you know, that much water available, it's --

CHAIRMAN STETKAR: Yes.

MR. STUBBS: I, sorry I can't give you a much better answer.

CHAIRMAN STETKAR: No, no that's fine. At least as I said, you at least pointed us to the available guidance in the Standard Review Plan and Branch Technical Position that basically governs your scope of review for this.

Okay. I don't think, I can't ask, I mean I've kind of raised the concern in my mind. I don't think there's much else that we can discuss on it. I did want to cut you off.

I still do want to discuss the break exclusion area out in the steam and feed water pipe area. The reason I wanted to treat that as separate, is that those areas just by the nature of the equipment that are located in those piping areas, they're less susceptible to these operation and maintenance type flooding activities.

I mean, yes, we got the main steam isolation valves out there and things like that, but you don't have the pumps and the heat exchangers and the valve connections in those areas that we do in the basement of the reactor building.

So unless there are other questions regarding the moderate energy pipe flooding in the reactor building, then we can start discussing the break exclusion area.

MS. LI: You want me to --

CHAIRMAN STETKAR: Yes, I mean as long as you're, as long as we have your, have you here.

MS. LI: Branch, as I say, Branch Technical Position 3-4 provides staff guideline of location of the postulate pipe failure, and it kind of follow the ASME design criteria. That one is the stress, one is fatigue usage factor. The built in safety margin that bring down the threshold.

So if the applicant can follow those guidelines and determining where are the breaks. But the speculation blank out a region that, some where people call it breaks exclusion zone.

That's the, in a general design would be between the inner isolation valve, or outer isolation valve, the component penetration.

CHAIRMAN STETKAR: That's where I've normally seen it, but those distances are typically also very short.

MS. LI: Yes.

CHAIRMAN STETKAR: And you get additional confidence because of the support at the penetration, plus the supports, the typical supports for the isolation valve.

MS. LI: Yes.

CHAIRMAN STETKAR: I don't know anything about stress analysis, that's why Pete's here, but I can at least see that. The difference in this plan is that the, in the main feed water and steam piping area, that break exclusion area extends for, as best as I could tell looking at the drawing, about 70 feet, which is a long distance.

And I was curious whether the staff, have you seen any other applicants who have done a similar type of analysis?

MS. LI: No. Some operation trend, I saw it, but same as you, when I first review it, and raised my eyebrow.

CHAIRMAN STETKAR: Yes.

MS. LI: And we asked RAI, you know, explain why you are apply the break exclusion region to, I think

the area you of concern is the main steam going --

CHAIRMAN STETKAR: Yes, it's between the, it's the --

(Crosstalk)

CHAIRMAN STETKAR: -- the penetration after the turbine building, almost to the turbine building, the wall.

MS. LI: Yes.

CHAIRMAN STETKAR: That's right.

MS. LI: Okay. Now so I kind of expect. So in Branch Technical Position, the normal area that we call break exclusion zone, it has eight additional design guidelines.

So you know, the applicant has to demonstrate that they are, meet the intent of those eight additional guidelines, which improve low stress, low fatigue usage factor and minimum the lengths of that piping, and avoid welding and also perform 100 percent volumetric inspection. So you know, and so on, so the applicant has to demonstrate that they meet those guidelines.

CHAIRMAN STETKAR: Yes.

MS. LI: But so, I have seen a lot of thing is that, you know, theoretically talking about the breaks exclusion is really from the component isolation, in fact

to the outer isolation valve, however one has to think of, okay, so also the outer isolation valve, so if it continues and if there's a break to the valve, then the consequence of that break to the operator of this plant.

So in a general design there will be either an anchor or a five-way restraint that near the outer isolation valve, which is really to protect the outer --

CHAIRMAN STETKAR: But in this case that five-way restraint is at the turbine --

MS. LI: It's near the wall.

CHAIRMAN STETKAR: -- at the turbine building wall.

MS. LI: Right. Yes.

CHAIRMAN STETKAR: As is the out buildings between the isolation valve of course, so --

MS. LI: I agree.

CHAIRMAN STETKAR: So we still have, pick a number, 65 feet of piping.

MS. LI: Right, but the, say one thing is as staff, we cannot detect how the new tenant, how applicant going to design their plant, so I ask RAI and follow up RAI, what I make sure is they indeed meet those eight additional design guideline.

CHAIRMAN STETKAR: Okay.

MS. LI: And make a statement to the best

practical, minimize the lengths of that piping. And also the agreement is that they don't patch that break for main steam and feed water line by any branch lines that come from those designs, break has to be postulated.

CHAIRMAN STETKAR: Yes. And they did that.

MS. LI: Okay. And I think there is another thing. I get their agreement is -- oh, right, there is a sister Branch Technical Position 3-3 which is the one I think this morning you also mentioned. That one starts with that.

CHAIRMAN STETKAR: Right.

MS. LI: So just originally BTP does not have that. So we asked the applicant to make that component, postulate that one square foot. Not make --

CHAIRMAN STETKAR: It's a longitudinal, yes.

MS. LI: -- approach, I mean the break, and so they were evaluate the longitudinal effects, including the pressurization that you talk about this morning.

CHAIRMAN STETKAR: Yes.

MS. LI: So with all those commitment and efficiencies they're reaching, the staff determine that their future of design is acceptable from the break

exclusion aspect.

CHAIRMAN STETKAR: Okay. Thank you, that's --

MS. LI: Like I say, I close my eyes.

CHAIRMAN STETKAR: No that's, and I understand what you did. I don't, not remember, I didn't look up all of the criteria, but thank you.

MEMBER RICCARDELLA: Excuse me did you say one of the criteria was no welding? One of the eight criteria?

MS. LI: To minimize the welding.

MEMBER RICCARDELLA: Oh, to minimize the welding. I was going to say, at 65 feet and no welding?

CHAIRMAN STETKAR: Sixty feet, yes in the in the steam and feed water pipe it's going to be tough.

MS. LI: Because we require 100 percent volumetric examination of the weld, so you know, they have to minimize that otherwise it would be difficult.

MEMBER RICCARDELLA: And is that also part of the in-service inspection program, 100 percent?

MS. LI: Yes. But is augment from the normal in-service inspection.

CHAIRMAN STETKAR: Thank you, that was, that helped a lot, helped me a lot anyway. Thank you. I do as I said, I kind of view that area in my mind

differently than I do the reactor building just because of the types of equipment.

There really isn't the same type of people interaction to that, so it does relate a little bit more on the design and inspection of the materials themselves in that area in particular. So thank you.

MR. GALVIN: Okay, next on the 3.4.2, this is the flooding on a structural design. The staff requested MHI to clarify the use of coefficient of friction of 0.7 at the soil-concrete interface. It's, this coefficient approach is also involved at the site of the construction analysis and it's being reviewed as part of that review.

The next point is in, this is actually related to Section 3.5.1.3 the turbine missile. So when you, when we get done with this, you can ask your statements or questions, after the staff's coming up.

MHI had acceptable ITAAC for turbine generator arrangement and turbine missile probability in DCD Revision 2. When they went and sort of updated their ITAAC, the staff had issues with their ITAAC. MHI has now proposed to modify their ITAAC and they've added Tier 2 information, and related to their reports.

And so what addition wording and a discussion of the turbine generator and determine

missile probability in response to RAI. And so the staff has found the proposed ITAAC and the Tier 2 information is an alternative approach to having the information in Tier 1. This is going to be a confirmatory item for Revision 5.

CHAIRMAN STETKAR: I think on the turbine missiles, I'll ask the staff. As I said this morning I read, because I'm not a materials person, I read in the MUAP whatever it is, 07028 which is more materials oriented, but I paid more attention to the over-speed analysis in MUAP-07029.

And as I mentioned this morning, I understand what the scope of that analysis covered and I had a couple of questions about the development of the failure rate that was used for some of the valves in there based on the zero failure evidence. But that's not the most important part of my concern.

The important part of my concern was the fact that the analyses seemed to omit everything from the turbine speed sensors up to and including the solenoid valves that drain the hydraulic fluid, such that the stop valves and control valves will close.

And I didn't, as I went through the SER, I didn't see any evidence that the staff looked at those types of issues. So I'm really curious whether, did you

look at the whole design of the system, all the way from the speed sensors through the digital signal processing logic, if I can put it that way, out to, including the hydraulic dump valves and then the valves themselves that are addressed in that technical report?

MR. GALVIN: We have John Honcharik, from DE to kind of start the discussion.

MR. HONCHARIK: Hi, I'm John Honcharik and I'll try to address some of those questions. The, that report, the 029, which is the turbine valve failure frequency. Basically that report is to evaluate and to determine inspection, and do our testing frequency for the valves, okay.

And that's based on the experience that they had in the valves and the design. And we did look at that. Also they do have some of the instrumentation there, you know, they do have the valves, but they also have the stop valve control system involved and so on --

CHAIRMAN STETKAR: That's right, but it's my understanding that's literally the, if I bought a stop valve, you know, it comes with a valve assembly and things that you hook up to it. And these are the controls that, for example on a control valve, the port, the hydraulic fluid, you know, for that valve, position it. It can't be the controls, my point is it can't be the controls I'm

concerned with, because those controls do not affect the valve individually. They affect the whole system.

MR. HONCHARIK: Okay, so you're talking like the speed sensor on the turbine borders --

CHAIRMAN STETKAR: I'm talking about literally the speed sensors on the turbine, the digital signal processing from those speed sensor signals to create a signal to trip the turbine, the output signals from that logic that tell, there's a set, I can't remember because I didn't pull up the hydraulic drawing, and it's a little bit complicated as most of them are.

But there are sets of solenoid valves that drain the hydraulic fluid from all of the turbine stop valves and control valves. And the solenoid valves in there, they're single failure proof, such that a single failure will not prevent a turbine trip, nor will a single spurious operation cause a turbine trip.

So it's like a, I can't remember whether it's one out of two, taken twice, type of hydraulic arrangement, but there are several. There are at least eight when I looked at the design, there are at least eight combinations of failures of two of those hydraulic fluid drain valves that will prevent the turbine from tripping.

And they cannot be associated with the

single turbine stop valve and control valve. They cannot logically be associated with whatever is in the stop valve or control valve signal processing or operation equipment because that literally can affect only a single valve in the model. A single stop valve, or a single control valve.

MR. HONCHARIK: Okay.

CHAIRMAN STETKAR: So I understand, and I'm concerned about a testing frequency for the integrated system so that we have assurance that you don't get an over-speed failure.

MR. HONCHARIK: Right. And maybe I should give a little background to the testing and inspection for it. In the SRP for the Section 10.2.3, there is criteria there that basically they have to meet. It says, well actually this is SRP 3.5.1.3, and basically they have to meet the ten to minus four, ten to minus five, but there is also instance where if they do not have a turbine missile analysis, okay, they must do the following. Okay.

And it basically prescribes what testing and inspection is required. And that basically is inspecting the turbine rotor every other outage, inspecting every valve every other outage. At least once a week do all valve testing, and at least once a

month, do the, each component of the governing system. Basically the speed sensors and all that, which you're discussing now.

So basically this analysis, you know, there's the one, 028, which is for the turbine rotor. They do that analysis in order to say that, okay, instead of doing it every three years, they can provide, provided that they meet the ten to minus four, ten to minus five, based on their orientation, that they can do this different inspection info.

CHAIRMAN STETKAR: According to them, it's now 20 years, right, which for them is now 20 years, if I remember correctly, right?

MR. HONCHARIK: Well the analysis says it's good up to 20, but they're going to do it every ten years.

CHAIRMAN STETKAR: Okay.

MR. HONCHARIK: Now, they also did it for 029, for the valve testing frequency.

CHAIRMAN STETKAR: Right.

MR. HONCHARIK: Okay, and this is only for the testing of the valves. Okay, and so they used all their information for the valves, and to demonstrate that they can do it quarterly instead of weekly, okay. But they're still going to do the monthly testing for the governing system. So basically that's the rate that is

set into the equation.

CHAIRMAN STETKAR: Okay, that's an important piece of information but I'll still ask you why do I have confidence that monthly testing of that system gives me less than ten to the minus five, over-speed failure? Because in my experience, I've looked at a lot of these systems and very, it's very unusual to find that two of them are at all similar.

MR. HONCHARIK: Right, okay. Yes, I'm not an electronics engineer for the system --

CHAIRMAN STETKAR: No, no, I'm not talking so much about the electronics, I'm talking about the Rube Goldberg fluid systems, the hydraulic stuff.

MR. HONCHARIK: Right, yes. We're also, you know, in the DCD under I think it's 10.2 they do talk about the hydraulic systems and everything. What they do to improve it, they use stainless and they have chemicals that they add to the lube oil and such that prevent these previous experiences they had and so on, you know, for that.

CHAIRMAN STETKAR: I understand that. I come back to the fact that if the failure rate for solenoid operated hydraulic dump valves is high enough, a monthly testing frequency will not give me assurance that the turbine trip failure rate is less than ten to

the minus five, for the over-speed trip failure rate. It will not.

I don't know what the failure of those valves are, because they've not been evaluated. If it's high enough, you might require a weekly testing frequency because of the specific design of their system.

This is not a generic one size fits all, it's their system. It is their design, with their electronic signals, and their particular configuration of M out of N logic, of little solenoid operated valves that need to work.

And I guess if you're saying that they're going to test that system once a month, I'm asking how do I have assurance that the once per month testing, of that checked part of the system, will indeed provide assurance that the overall integrated frequency of over-speed failures is still less than one times 10 to the minus five per year? I haven't seen an analysis that gives me that confidence.

MR. HONCHARIK: Okay.

CHAIRMAN STETKAR: Because it's not include in the 07029.

MR. HONCHARIK: No, and that would not. That's basically the valve test frequency. But there might be some reliability program that they have for

their over-speed system, but I'd have to defer that to the systems speaking for that, electronics okay?

CHAIRMAN STETKAR: Well you said they have committed to a monthly testing of --

MR. HONCHARIK: Yes, that's part of their inspection program --

CHAIRMAN STETKAR: -- this other program.

MR. HONCHARIK: -- in the DCD.

MR. GALVIN: In Chapter 10 you said, right?

MR. HONCHARIK: Yes, in Chapter 10, 10.2. I think it's on the table.

CHAIRMAN STETKAR: Okay, I'll go look at that. But still even if it's a month, my whole point is that, that month, the effectiveness of even that monthly testing depends on the particular configuration of the valves that they have.

How many series in parallel flow paths, basically, and the reliability of those valves. And because people do the reliability calculation using a, what I call a stand-by failure rate model, which means that the reliability of the valve is directly proportional to the test interval.

It's not clear to me, that with this particular design, a monthly test interval would give you that assurance. It might, it's just I haven't seen an

evaluation to confirm that. You said Section 10.2 has, where's that --

MR. HONCHARIK: Yes.

CHAIRMAN STETKAR: Okay thanks.

(Pause)

CHAIRMAN STETKAR: Anything else on turbine trip? I just get frustrated by this because, it's, I've yet to see any applicant or any design come in with a true end-to-end analysis, from the sensors all the way out to stop valves and control valves. So this is not unique to this application by the way. Thank you.

MR. GALVIN: Section 3.5.3, design of missile barrier. The staff proposed to MHI to provide an analysis assessing the local effects of an automobile missile on all seismic Category I structures not covered by Reg guide 1.76.

The staff is reviewing the applicant's response. If you had any additional questions on Sections 3.5 or 3.6 probably is now the time.

CHAIRMAN STETKAR: I don't. Any of the other members?

MR. GALVIN: Okay, we'll go ahead to 3.9 now. The next two questions are associated with, as you've heard this morning, getting additional information related to the design of the steam generators

in response to the failure mechanism, San Onofre is the first one.

The staff did have some interactions with the applicant, but we haven't received their official response. And again this is again additional information on designs of the US-APWR steam generator, and we're awaiting their response.

It talks about the topics we asked about. You know, the, design of the steam generator tube bundle and the design criteria for the steam generator tubes, retainer bars against flow-induced excitations, including random turbulence, fluid elastic instability and vortex shedding. We're still interacting on that.

Section 3.9.3, staff reviewed the seismic analysis of those major components. Supports was impacted by MHI's seismic analysis changes. The status is staff found the analysis methods in MHI's latest technical reports 10006 and MUAP-09002 acceptable.

Pending the incorporation of the MUAP-09002, Revision 3, this issue is a confirmatory item. So the staff looked at the analysis and the latest revision, and found it acceptable. There's a another revision coming in December, or it's scheduled for December, but that doesn't address this area.

3.9.4.1, those three questions, again it's,

there was limited margin between the calculated control rod drive mechanism deflection and the design limit prior to seismic analysis changes.

The staff is awaiting the updated maximum control drive mechanism deflection based on the updated seismic analysis. We haven't received this yet. I think we received partial, but we haven't received all the responses yet.

3.9.6, this has involved the, MHI has chosen to address implementation of the functional design, qualification, and in-service testing programs in accordance with the requirements of 10 CFR 52.79, therefore make available for audit the sample of applicable design or procurement specifications. And this says the open item will be closed upon conducting the audit, which is not currently scheduled.

ASME allow of the design cert can be support to determine specification audit, in this design center it's the design certification Op. And he's going to support the audit, and we're, haven't, we don't have a schedule for that at this point.

Section 3.10, staff requested MHI to provide a list of components of the gas turbine generator system to be qualified and a description of the methods, criteria, and procedures. This information is included

in technical report MUAP-10023, which was impacted by MHI seismic analysis changes.

The staff is reviewing a revised RAI response, revision to MUAP-10023. We've had some discussions and we're awaiting for the final revision. We may have additional interaction before that's cemented.

Number 3.11, MHI's commercial-grade dedication process does not provide an alternative means of environmental qualification since it does not address all environmental qualification requirements specified in 10 CFR 49 for electrical equipment and the guidelines of ASME QME-1-2007, Appendix QR-B for non-metallic components of mechanical equipment.

The staff is reviewing the applicant's response. The commercial-grade dedication process and environmental qualification, are really are two different things, that this, they're sort mixing them too, and we're trying to sort that out.

These next three are associated with the radiation protection aspects of, radiation dose aspects of environmental qualification. Staff has concerns associated with the calculational methods and results for total integrated dose to equipment inside containment following loss of coolant accident.

Actually the applicant has given us some information, but because of other priorities, we haven't reviewed it yet. The next one, the staff has concerns associated with the calculational methods and results for the beta ray source term for equipment inside containment following a loss of coolant accident.

And the staff has concerns associated with the inconsistencies in the operability times of post-accident equipment inside containment. And so this is associated with identifying what equipment is PAM. We have received a response on that, but that's still under review, post-accident monitoring.

MEMBER SCHULTZ: So Dennis, we didn't discuss those in the applicant's presentation. Is it Chapter 12, open items, are related to post-accident?

MR. GALVIN: Actually I don't, this slide probably could be maybe wrote a little better. Post-accident monitoring is actually Chapter 7, there's, you know, what's the doses where the, there was some question about how long the equipment needed to be available.

And of course you have to decide what equipment you need before you can decide what is needed to be available, and they haven't got that far. We found some, I guess, Ron, you want to explain

further?

MR. LAVERA: Good afternoon, is this on?

MR. GALVIN: Yes, yes it is.

MR. LAVERA: I'm Ron Lavera, I'm the Chapter 12 Radiation Protection reviewer, and I also do the radiation protect, radiation portion of the Chapter 3.11 review. When doing the review for the equipment, what I'm looking at is the equipment that's identified in Table 3D-2, in Section 3.11.

But when I do that, I also go and look in the other sections of the DCD, in particular, Chapter 7, to make sure that there is a correlation between the instrumentation that's described in Chapter 7 versus 3D-2.

So if there is, the equipment that's described in Chapter 7's, got to match the equipment that's described in Chapter 3.11's, got to match the equipment that's described in Chapter 15. And also has to match what's in Chapter 16. So if there isn't this one-to-one correlation, then you get this, I have a question regarding what's going on. So that's why it's tied to the Chapter 7 REI.

In return, what you also have, is if you have a piece of equipment that's identified that you may need to go access, following the accident, then that involved

a mission dose, which is now described in Chapter 12.4. So I'm looking at this thing as a whole integrated thing. So it's all got to line up, and where it doesn't, questions have been issued.

MEMBER SCHULTZ: All right. Thank you for the explanation. That's clear, thank you.

CHAIRMAN STETKAR: And as part of that, just to make sure I understand it, in terms of this Section 3.11, the concern is questions about the assessed operability times for that scope of equipment. Is that right?

MR. LAVERA: I'm not sure I'm understanding your question.

MR. GALVIN: Well some of those, at one point there was a question of, they had an operability time of one duration, and then an operability of another duration in a different section, right?

MR. LAVERA: Yes, understand that I'm not going in to analyzing whether the operable duration that they are coming up with, is satisfactory, but where they come up with and operable duration in one place, it needs to match the other.

CHAIRMAN STETKAR: You want to make sure that it's consistent.

MR. LAVERA: That's the limit of my review.

CHAIRMAN STETKAR: Okay.

MR. GALVIN: The next one is 3.11.42, the staff requested MHI to provide an equipment qualification data package template. Applicant has provided an equipment qualification package template. Staff has provided some feedback, and we're awaiting additional information from MHI.

The next one, MHI has not demonstrated how the US-APWR satisfies the environmental qualification requirement for electrical equipment, 10 CFR 50.49 with regard to the treatment of non-safety-related electrical equipment located in a harsh environment, whose failure under postulated environmental conditions prevent satisfactory accomplishment of safety functions, and by deleting "important to safety" words throughout environmental qualification documents.

Now regarding number 1, Peter might want to assist me here. I think we have looked at, they have gone through and done analysis and determined that they don't have any non-safety equipment that could impact the operability of the function of a safety equipment. Is that correct, Peter?

MR. KANG: The applicant, only one of them respond, I have their response, what they submitted to us was that they said that they have evaluated most of

the non-safety, 10 CFR 50.49 and my own qualification, the requirement is, involves the two sections, in other words the safety-related, and the non-safety- related which is valuable for the fact of safety- related equipment.

And we were asked the question about whether non-safety-related exempt, do you have any, have you ever looked at it, or do you have any non-safety-related equipment which value can affect the safety, operational safety of the equipment.

Basically, MHI in their response indicated they have evaluated, they have not found any. And in fact they even put it in that statement in the DCD. That's the first question. And the second is the deleting "important to safety" words. If you will, we consider this is a sort of pretty important words.

You, in words, because that is a sort of general implies all the equipment basically. It's in the, for to begin with, it is a regulatory word. And they just cannot select this word and delete it everywhere.

And as I said some basic reason why they done that, but staff doesn't agree with that. So we are sort of had a few conference calls, and to revert back to leaving important safety words in.

MR. GALVIN: We consider this a standard

discussion at the time.

MR. KANG: Yes, this is a normal --

MR. GALVIN: We've had some interactions and, you know, we're still interacting with the MHI on the specific language to address, so that it's clear that 10 CFR 50.49 requirements are within scope and not, you know, they want to use some alternative words, and we're still working on that.

CHAIRMAN STETKAR: The, I'll let you finish the last couple of bullets on here.

MR. GALVIN: Okay.

CHAIRMAN STETKAR: Because I had a couple of other more general questions, but we'll get through the specifics first.

MR. GALVIN: Question 3.11.55, the ITAAC do not include demonstration of environmental qualification of non-metallic parts of mechanical equipment. They have a note on the table that sort of excludes things, and the note's not very -- The note's problematic and we're working out, on the Tier 1 table, so we're working on it. Whether the note's needed or it needs to be refined. And we're still interacting with the applicant on that.

And then 3.12, design loads for piping were updated because of changes to MHI's seismic analysis

methods. In addition, the seismic analysis methods of steam generator supports were unclear.

Based on a review of MUAP-09002, Revision 3, the staff has found the piping analysis modeling acceptable and the treatment of the steam generator non-linear supports conservative.

Now MHI has informed that they plan on revising the response, so I guess we're going to have to take a look at that. That's more recent than what we've got here. And that's it.

CHAIRMAN STETKAR: Okay, I had a couple general questions. One of them does relate to this notion of, important to safety, versus safety-related. And it, I read the SER, there's discussion not only on the environmental qualification, but it also throughout the SER, it sort of touches on this issue.

And it's not clear to me yet, from the staff's perspective. So what I'd like to ask is, we do have in the DCD, there is that Table 17, whatever it is, .4-1, that lists non-safety-related equipment that is quote, unquote, "risk significant", or it's considered to be important enough to be included in the Design Reliability Assurance Program.

From the staff's perspective, does that table then define the scope of equipment that is quote,

unquote, "important to safety", or are other criteria invoked? And if there are other criteria, I'd like to better understand what they are.

(Crosstalk)

CHAIRMAN STETKAR: It can't, as you said, it can't be sort of on a case by case basis, section by section of the SRP, and negotiated consensus of what is the equipment that satisfies those words. Because you can very quickly lead to inconsistencies that way.

That something is treated one way in one part of the review, and a different way in a different part of the review. And yet, it is the same piece of equipment that provides some safety function.

MR. KANG: In the order he put in qualifications. In other words that 10 CFR 50.49 rule is pretty distinct, in word of safety. Word is very written there. In other words it is nothing to be related with the risk, nothing with the risk, tied with the risk.

CHAIRMAN STETKAR: Right.

MR. KANG: So that's why we were in a way, we like to compliance with the regulations. We'll like to see use that words. That's why we are insisting this is a regulatory words, but the way MHI's approaching is, approach is, they use the segment of a words, the guidance

word from 3, SRP, 3.11, which is an SRPs are written for staff, for the staff.

Not for the applicants, applicants get, should be able, should tell us how they meet 50.49. But on the other hand, they use that guidance word from 3 SRP 3.11, and here you are. We are considering --

MR. GALVIN: Peter I think you're --

(Crosstalk)

MR. GALVIN: -- question.

CHAIRMAN STETKAR: Let me just step back from this, because my, I'm not as familiar as you are with all of the rules, nor the Branch Technical Positions, nor the regulatory guidance. I'm just sort of naively think of the world as I have one set of equipment in the plant that's given the term, safety- related. And I have another set of equipment in the plant now, that is not safety-related, but is determined, deemed to be, important to safety. And then I have a third category of equipment that is not safety-related, and not important to safety.

And my question is for that middle category of equipment, that is not safety-related, doesn't fall in the first box, but it does fall into the second box, it is quote, unquote, "important to safety". How does the staff and the applicant consistently identify that

set of equipment?

Such that if I'm looking at EQ, or I'm looking at flooding, or I'm looking at any of the other parts of the review that we look at, I understand clearly between the staff and the applicant how a particular piece of equipment is categorized. What box it appears in? And that's my basic question.

MR. GALVIN: All right, I think that Theresa --

CHAIRMAN STETKAR: And so that there's a clear understanding from both sides how that particular piece of equipment will be treated.

MR. GALVIN: Theresa Clark did the Mechanical Engineering Branch Chief, I think has some comments.

MS. CLARK: Hi, I'm Theresa Clark as Dennis said, also formally from PRA, so I kind of feel that side of it as well. I do very much sympathize with the frustration that you're expressing. I understand the concern, and I think we could probably also have some fruitful off-line conversations about this --

CHAIRMAN STETKAR: Sure.

MS. CLARK: -- even separate from the APWR discussion. There, while I understand your desire for a universal definition, it has been dealt with on a very

case by case basis, in different sections.

You know, the one that Peter referred to, I was furiously opening up on my phone, because it's very interesting actually. In 50.49 the equipment that's important to safety from a 50.49 perspective, is explicitly defined in the rule as which things are important to safety from that perspective.

CHAIRMAN STETKAR: Okay.

MS. CLARK: And so that's actually very interesting to look at. In other venues, such as GDC 1, which has to do with quality assurance, commensurate with the importance to safety of equipment. That has been less well defined and there has been controversy over, you know, probably 30 years about how that's been defined with commission interaction as well. And we didn't end up with a universal definition that could be easily applied.

However when you look at different review areas such as, you know, you look at the seismic classification which we spoke about this morning, the quality here at classification, Reliability Assurance Program, while they're not necessarily unified, we do try to make the criteria for classification within those areas as clear as possible, so that you can find all of them.

Unfortunately, I think at this point, that's probably the best kind of overall explanation I can give you, because we can't, we don't have that universal definition and it is something that staff is considering. But we found ways to get what we need to do to make the safety findings under the GDC.

MR. HICKS: Excuse me, this, Tom Hicks from MNES, do I just --

CHAIRMAN STETKAR: Absolutely.

MR. HICKS: I think we agree exactly, what she just said. And that is that, that term is defined differently in different places. And that is the reason why we did not put the term in 3.11 of the DCD, was because we felt people would misunderstand what it meant.

So what we did, was we took the scope that's defined in the SRP, A through F, it says here's the equipment that's included in this program. And it's very clear what that equipment is. And that scope is also listed in Reg Guide 1.206. Okay.

So that's what we did, and it's very clear what equipment scope is in there. And we felt like if we put the words back in, important to safety, it's going to do exactly what she just said, it's going to confuse people because it means different things in different parts of the regulations. And so that's why we don't

want to put that word in there, those words.

CHAIRMAN STETKAR: We'll let the staff and the applicant discuss that within the specific context of the EQ scope. This issue has come up, in some sense, you're right. This is more of a generic issue. We've raised it pretty much in every design certification that we've been involved in. And this one is no different in that sense.

What I'm trying to, what is specific to this design center, which is a little bit of what I was trying to get a sense of, is this design center does have a table, that's that 17.4-1 that identifies a list, it's a very large list of structured systems and components that are deemed to be important to safety.

Well they use the term, risk significant, and they're significant enough to risk, that additional reliability assurance and additional quality controls are warranted.

And what I'm trying to understand from the staff's perspective, I think I'm, is whether or not for your purposes in the review, does that list define the non-safety-related equipment that is quote, unquote "important to safety"?

What I'm hearing is, no, not necessarily. So that's from this perspective. That at least clarifies

my continuing confusion.

MS. CLARK: And again I sympathize with that --

CHAIRMAN STETKAR: What I'm, on a generic sense, you've discussed it internally and I think would probably want to pursue ACRS in a different venue, you know, this whole notion. You've had some discussions in the past.

MS. CLARK: And I think you've been in on those discussions. I think, you know, there is not an official position that I'm aware of that says, that table equals important to safety.

CHAIRMAN STETKAR: Okay. But I do see discussion in other parts of the SER, I can quote references to that table as part of the staff's questioning about why isn't this particular set of equipment identified in a particular quality group, or a particular seismic group, and things like that.

MS. CLARK: We do --

CHAIRMAN STETKAR: So the staff does seem to refer to that table, you know, in some areas of the review.

MS. CLARK: We do use information like that table to provide insights to our staff, to say these are more important areas that you might want to look at. But

in terms of an official definition of, important to safety, we haven't gotten there.

CHAIRMAN STETKAR: Okay. Thank you. That, helps, thanks. Let me just make a note here.

(Pause)

CHAIRMAN STETKAR: The other question that I had, kind of a general question, and again if you were here this morning you heard me ask the applicant about the in-service testing frequencies for motor operated valves.

And their justification for using a what is effectively a two year testing interval versus a quarterly testing interval for a fairly moderate, let's call it, subset of all of the valves in the plant.

And I was curious how the staff reviews those testing intervals and what sort of thought process do you use? Because I, when I initially read the words in the table that says, well if we test this valve it will disrupt a normally operating required support system. I said, oh, that sounds quite reasonable until I thought about the fact that it isn't always operating.

And indeed the way people, to satisfy the pump testing requirements, and other normal operational requirements, it's not clear to me why they could not invoke a quarterly testing interval for a large number

of valves.

MR. SCARBROUGH: This is Tom Scarbrough, I'm in Mechanical Branch. In this particular case for US-APWR, we had Pacific Northwest Laboratories who has IST permit review experience, go through the entire table in all their systems.

And if you look at the SER you'll see a lot of places where it was in the RAI saying why weren't all these valves tested quarterly? Why are they pushed back?

And so they went through a whole process and where they found a concern they raised it, and we interacted with them, and they wrote a number of changes that you saw on the tables over the years, to move things from cold check out, or a peeling out to quarterly as they negotiated, you know, could they do this a quarterly test?

Now the ones that I understand that you were asking about, were the safety depressurization valves. And there was a block valve that --

CHAIRMAN STETKAR: Were you here this morning, Tom?

MR. SCARBROUGH: Yes.

CHAIRMAN STETKAR: Okay. Yes, I mean there was a couple. There's sort of two categories, one

is that type of valve. The other is valves in normally operating support systems, service water, component cooling water, and that forth, and they're different rationales for both of those. So let's address the SDVs.

MR. SCARBROUGH: Yes, now this one has to do with similar to what was decided with AP1000, where they had two AVBs, automatic depressurization valves in series, and they're both closed. The rationale here is that the applicant did not want to be in a situation where a mispositioning event would create a LOCA.

And when, it went through, this is the review process we had. So even though they have a valve, but they closed the other valve, but opened it, they did not want to get in a situation where an inadvertent operation would throw them into a LOCA.

CHAIRMAN STETKAR: Okay.

MR. SCARBROUGH: And historically, we've been receptive to that argument. So that's was the basis for why these safety depressurization valves, whereas the block valves which is normally open anyway, so we can, it can go up and down quarterly since it's already open.

But the safety depressurization valves themselves that are normally closed, they put in their table that they would only do it now because they do not want to have an inadvertent event cause them to have a

LOCA. And that was the same position we had with AP1000 and we consider that to be reasonable.

CHAIRMAN STETKAR: Okay. Let me just make a note and I understand that rationale.

(Pause)

CHAIRMAN STETKAR: I'm a slow writer, I used to be able to write more quickly and legibly. I can't do that anymore, because I have a computer, but I type really slowly, so, yes. And I understand that rationale, I kind of get it.

What about the other normally operating systems, the ones where they say, well if we, let me give you a specific example if I can find it here quickly. I have a lot of notes here, it takes me a while to find it.

Okay. A good example, one example you may want to look into, this one I couldn't find, I couldn't figure out for the life of me. And this is not the specific concern that I eluded to earlier. Is that the safety injection pump suction isolation valves, the SIS MOV-001 ABCD, according to the table, are stroke tested during cold shut down.

That's a stand-by system. The valves are normally open. The containment spray RHR pump suction isolation valve CSS MOV-001 ABCD, are stroke tested once a quarter. I can't for the life of me figure out why the

SI pump suction valves can't be tested once a quarter.

MR. SCARBROUGH: Okay.

CHAIRMAN STETKAR: There is no rationale for why not quarterly test them. They're simply blank over there. It may be an oversight.

MR. SCARBROUGH: Okay, we will check that.

CHAIRMAN STETKAR: But not to the point because this is more germane to the systems that I found kind of consistent rationale. If I look at component cooling water, return and supply isolation valves, and if you want to write down the valve numbers they are NCS MOV-007 ABCD, those are the return valves. They're the return from each header to the pumps. And the corresponding supply isolation valves NCS MOV-020 ABCD.

The frequency of stroking those valves is once per refueling outage, or essentially once every two years. And there's a Note 7, that applies for the valves. And this Note 7, is used quite frequently throughout that table.

And it says, exercising these valves would stop seal injection return water, or cooling water of the reactor coolant pumps. Such stop of the water may result in damage to the reactor coolant pump or reactor trip. These valves are exercised during cold shut downs and these components do not require water flow.

I will grant you that for the two trains that are normally running. The two trains that are in stand-by could cycle those valves or leave them closed and it doesn't affect cooling.

MR. SCARBROUGH: Yes.

CHAIRMAN STETKAR: During normal operation only two trains are running, two are in stand-by.

MR. SCARBROUGH: Right, that's good. We'll check into it.

CHAIRMAN STETKAR: And that applies, that similar type, there's a Note 7 that says that, it's related to interruption of cooling, for the record --

MR. SCARBROUGH: Again we --

CHAIRMAN STETKAR: -- cooling cups. I'll give you your turn. And there's a Note 6, that applies to other systems, but it's the same basic notion. It says, exercising these valves would stop necessary line for operation of such as utilities et cetera, therefore exercise testing will be performed at cold shut down.

That note tends to apply to things like essential chilled water, essential service water, cross-tie valves, among non-essential and essential chilled water systems and so forth.

There too, during normal operation only two

trains are operating, two are in stand-by and I couldn't for the life of me figure out why you can't cycle the valves for the stand-by trains.

MEMBER RAY: Well are they heat exchangers though? Presume they only had a single line, in other words two, they have two separate heat exchangers for the reactor pump seals?

CHAIRMAN STETKAR: No, this is the supply to, the component cooling water system on this plant is interesting. Let me just say that. How they get water to reactor --

(Crosstalk)

CHAIRMAN STETKAR: -- this is, these, the two valves that I'm talking about are the valves, think of them as the, it's easiest to think of them as the suction and the discharge valve of the component cooling water pump.

MEMBER RAY: No, I just was just trying to say --

CHAIRMAN STETKAR: It's not quite that simple, but it's --

MEMBER RAY: -- at some point they're separate, but at some point if you're talking about at the valves at the seals, that's not true. That's not true.

CHAIRMAN STETKAR: Pumps A and B, hate to do this but, pumps A and B supply header A-1, which then supplies reactor coolant pumps A and B.

MEMBER RAY: And you're not talking the valves.

CHAIRMAN STETKAR: I'm not talking about the valves --

MEMBER RAY: --off the header?

CHAIRMAN STETKAR: -- off the header. Those, that's clear. If I close that valve and they say you can't touch that valve during normal operations, and absolutely you can't touch that valve. But cycling, if pump A is running, cycling the valves for pump B has absolutely no affect whatsoever on component cooling water flow to anything.

MEMBER RAY: Okay, I got it, never mind.

MR. SCARBROUGH: Well these again, we'll followup.

CHAIRMAN STETKAR: And I don't know how the staff approaches that, because in some sense it's, the problem is that in some sense it depends on how the eventual plant operator will operate those systems.

But in another sense if I'm now inheriting this certified design, as a plant operator, and I only need to test those months, those valves once every two

years. That is the starting frequency that I use for my now, what will eventually become, a risk informed service testing program which will extend at, beyond those two years. And that's the concern that I have.

MR. SCARBROUGH: Right. Well actually the way we, this review is setup for this operational program, the design certification applicant is providing a description of the program.

CHAIRMAN STETKAR: Yes.

MR. SCARBROUGH: And then the COL applicant can incorporate by that right reference in the program.

CHAIRMAN STETKAR: Right.

MR. SCARBROUGH: But when they get to the point where they're completing their ITAAC, and we're actually performing operational program inspections, we will be looking at the program itself, including each of these justifications for cold shut down. And so at that point, is when they have to actually justify all of these extensions.

CHAIRMAN STETKAR: Right.

MR. SCARBROUGH: Because of right now, this is a description of the program based on their best knowledge. But when we get to the phase when they're building the plant and we're doing the operating program inspection, the staff has to accept full review.

Reviewing each of the justifications for acceptance.

So there is no finality at this point with the design certification. It really only comes at the COL licensing stage. Actually it's even during the, sort of the ITAAC process when we're conducting our operational program inspections. And that's when there will be finality.

You know, so as they build the plant, we'll be looking at each one of these justifications and saying, okay now that you built the plant, you can actually test this quarterly, the way you have the plant setup.

And then they would, you know, we would not accept the justification, because we have to accept their justification, or they have to go back to quarterly testing. So this is the best knowledge right now of what they think they can do. But it won't be final until we actually do the operational program inspection.

CHAIRMAN STETKAR: And there again, I'm a slow writer, that's, that's a good explanation. I also see that rationale. Of course, then it's incumbent, very incumbent on those inspectors to think about these types of issues.

MR. SCARBROUGH: That's right, and they do.

CHAIRMAN STETKAR: In some sense, you could

do the design certification a couple different ways. You put quarterly testing in there, and then they justify why you can't do it per quarter, or put do your testing in there, and rely on the inspectors to tell you that you need to do it quarterly.

MR. SCARBROUGH: And how we have it setup for Vogtle, getting ready for Vogtle this summer, was that the regional inspectors will be supported by NRO and the IST program reviewers. And they can assure that the justifications are acceptable.

CHAIRMAN STETKAR: Okay. That helps. That helps a lot.

MR. SCARBROUGH: Okay.

CHAIRMAN STETKAR: That helps a lot. Thank you. Do any of the other members have any questions for this staff on Chapter 3? If not, thank you.

MR. GALVIN: Thank you.

CHAIRMAN STETKAR: Now we face a little bit of a time management question. Ryan, we are at 4 o'clock, is? Okay. Let me ask Luminant, I haven't seen Luminant's, I have to be careful, because I'm still on the record here. I haven't seen Luminant's presentation, what, we have a closed session scheduled at 4 o'clock. It's 3:15 p.m. now, we need to take a break

here sometime.

What I'd like to do in the interest of time management is, if we can have Luminant start their presentation and run through about a half hour of it, then break for the closed session. And finish up Luminant's presentation, it's probably the best use of our time, unless that creates too many headaches for you.

MR. WOODLAN: No that sounds good.

CHAIRMAN STETKAR: You good with that? Okay, let's do this then so that we can rearrange seating and such. We'll take a 15 minute break, reconvene at 3:30 p.m. and have Luminant start their presentation on Chapter 3, and see how far we get through it before our 4 o'clock closed session. So we'll recess until 3:30 p.m.

(Whereupon, the foregoing matter went off the record at 3:14 p.m. and went back on the record at 3:31 p.m.)

CHAIRMAN STETKAR: We're back in session. This is still an open session. We're going to hear from Luminant on their Chapter 3, again except for Chapters, or Section 3.7, 3.8. For opening, I'll ask Stephen Monarque from the staff, do you have any opening remarks?

MR. MONARQUE: Thank you. Good afternoon. My name is Stephen Monarque. I am the lead Project

Manager for the review of the Comanche Peak COLA. I want to thank the committee members for the opportunity to present COLA Chapters 3 and 9 today and tomorrow.

Chapter 3 discussion will not include Sections 3.4, 3.7, 3.8 due to our ongoing review of hydrology and seismic analysis. And with that, I'll turn it over to Luminant.

CHAIRMAN STETKAR: Thank you.

MR. WOODLAN: Good afternoon. My name is Don Woodlan. I'm the Licensing Manager for the COLA, COLAs for Comanche Peak, Units 3 and 4. Pleasure to be here again, and be briefing the ACRS Subcommittee.

Before we start, let me briefly make as I mentioned on the US-APWR, their slow down obviously has an impact on us. We follow their work activity, and we are closely linked to their work.

Luminant has assessed our options on the review, and as we stated in our letter, which we have filed, that we have decided that it's not worth the resources at this point in time, I've informed them that, also the NRC staff, to continue the review of the Comanche Peak COLAs until we better understand and the reviews of the US-APWR are further along. Therefore as of March 31st of next year, we are going to fully suspend the ongoing reviews of our application.

Some of them were actually suspended earlier. We're working with the staff to identify those activities that we want to get completed, including a couple ACRS briefings if we can possibly get them done, to leave the product in the best condition we can, and it clear at division, so that when we do restart, there's a very obvious point to restart all the activities.

CHAIRMAN STETKAR: You say that's March 31st, is the --

MR. WOODLAN: That's the end date, yes.

CHAIRMAN STETKAR: That's the end date, so we'll need to over the next 4 months, carefully coordinate our schedules with your schedules, so that we make sure that whatever activities need to completed, at least from our perspective --

MR. WOODLAN: Yes.

CHAIRMAN STETKAR: -- we have those in our subcommittee schedule. And if necessary, our full committee. As you know, our full committee does not meet in January, so we'd be looking at February, March as our only full committee meetings.

MR. WOODLAN: And because we're getting this subcommittee meetings in, these two, that you'll probably really want to try and get those in.

CHAIRMAN STETKAR: Yes.

MR. WOODLAN: If we can.

CHAIRMAN STETKAR: Yes, yes. So we'll need to do that.

MR. WOODLAN: All right, with that let me move onto the presentations. We are doing Chapter 3 first. And the agenda is as shown here. We'll do an introduction, just a few good words about the license conditions in this chapter. And the we'll get into this site-specific aspects, which includes three SER open items. It's not as brief as some of them, where we were, IBRed all the way through.

We'll do some site-specific stuff here, but in general. Overall we're going to be IBRing. And that's covered here in the introduction. We're sticking to the same approach we have for our earlier chapters. The FSAR primarily uses the IBR methodology.

We have taken no departures in this chapter from the US-APWR DCD. All COL items that are in the DCD, have been addressed in the FSAR. We do have the three open items, which we're going to cover with the individual sections where they apply. We have five license conditions, and we have no contentions pending before the ASLB.

The next three slides cover the license condition of the, and I'm not going to go into them in

detail. You can read them if you want to, I know they're not a significant item as far as ACRS goes.

But several of these items do relate to things we talked about earlier, with the US-APWR. For example 3.1 on the pipe hazards analysis. And that is an analysis that will be done, in this case, prior to actually installing the pipes in any given area of the plant. We will complete the pipe hazards analysis for that.

Let me go to the next slide, in-service testing, now which we were just talking about with the US-APWR. And I do agree with the discussions that were held there. We will create our in-service test program.

It will be reviewed by the NRC via inspection activities. And we do have that first bullet there, 3.3 is where we provide a schedule to the NRC of our development of the IST and implementation. So that they can come out and do their inspections.

And this is where there is a, by the way, we agree and have proposed wording for all five, or all of these license conditions. There is a slight difference between our wording and the NRC's wording as they presented in the safety evaluation. And that we recognize there may be a period of time after we get our COL, before we start construction.

The current SE wording requires that we start providing this schedule to the NRC within 12 months of the COL. Our variation is, it's within 12 months of the COL, or upon starting construction, whichever comes later. Recognizing there may be a dead period there. It's kind of crazy to be filing reports every year for no work. We have worked that out with the NRC.

And this last one, is the last set of license condition, there is one license condition mentioned in the SE, relative to turbine inspection program. It's not a chapter 3 license condition, but it is related as we discussed this morning with the US-APWR.

With that, we'll start moving into the site-specific aspects of Chapter 3. And I'll turn it over to Mr. John Conly.

MR. CONLY: Thank you, Don. My name is John Conly. I'm the COLA Project Manager for Luminant. All Chapter 3 appendices, except 3 Delta and 3 Kilo are incorporated by reference without supplement or departure. Those two appendices 3 Delta and 3 Kilo will be discussed along with their related sections later in this presentation.

Section 3.1, conformance with GDC. In accordance with GDC 45, the essential service water system piping is arranged to permit access for

inspection. Manholes, hand holes or inspection ports are provided for periodic inspection of system components. The integrity of underground piping is demonstrated by pressure and functional tests.

Section 3.2, classification. The site-specific, safety-related systems and components that are designed to with stand the effects of earthquakes without loss of capability to perform their safety function, are identified in Table 3.2-201 as the essential service water system.

The ultimate heat sink, except the basin makeup piping and valves, and the ultimate heat sink essential service water pump house ventilation system. The DCD methods of equipment classification and seismic categorization are applied to Table 3.2-201, including, important to safety, and, risk significant, as discussed during the DCD presentation this morning.

Section 3.3, wind loadings. The site-specific, basic severe wind speed of 96 miles per hour, is enveloped by the standard plant severe wind speed of 155 miles per hour. There are no site-specific seismic Category 2 buildings or structures. There are no site features that promote wind channeling or buffeting that warrants special design consideration.

Site specific seismic Category 1

structures, that is the ultimate heat sink related structures, the essential service water pipe tunnel, and the power source fuel storage vaults are designed for the same tornado loadings and combined tornado effects using the same methods for qualification as the standard plant, SSCs.

Further, the design basis hurricane wind speed for site-specific seismic Category 1 structures is 145 mph as you will see in FSAR Revision 4 very soon. This is bounded by the DCD 160 mile per hour design basis hurricane wind speed.

CHAIRMAN STETKAR: John, just to, I know we haven't seen it yet, but was that 145 mile per hour wind speed evaluated according to the guidance in, and I've forgotten the Reg guide number, it's like 1.1.21 or 22, something like that.

MR. CONLY: Yes.

CHAIRMAN STETKAR: It is, okay.

MR. CONLY: Failure of non-safety-related buildings and structures on site will not jeopardize safety-related structures, systems or components. And will not generate missiles that are not bounded by FSAR Subsection 3.5.1.4.

Site specific or field routed safety-related structures, systems and components are

evaluated for the need for structural reinforcement and or missile barriers. Are there any additional questions? I will ask Todd Evans to continue with 3.4.

MR. EVANS: All right. Todd Evans with Luminant, and I'm going to cover Section 3.4 which pertains to flooding design. There are no site-specific flood protection measures for Comanche Peak 3 and 4 since the plant grade is above the design basis flood levels.

All seismic Category 1 structures that are below grade are protected against flooding, including effects of groundwater. And all site-specific safety-related structures which includes the ultimate heat sink related structures, the essential service water pipe tunnel, and the power source fuel storage vaults have been evaluated for internal flooding.

These structures have independent compartments for each safety train, such that internal flooding of any one compartment, will not prevent other trains from performing the required safety functions.

CHAIRMAN STETKAR: Do you have, and we may see it tomorrow, when I made a -- interesting -- I asked this morning, MHI, who has the responsibility for evaluation of flooding in essential service water pipe tunnel, the ESWPT, and the essential service water pipe chases, the ESWPC? And they said they belong to you.

MR. EVANS: And that's correct.

CHAIRMAN STETKAR: Okay. So, that third large bullet there, also includes the pipe chases, right?

MR. EVANS: That's correct.

CHAIRMAN STETKAR: And my problem is I don't know the exact configuration of the pipe tunnels and the pipe chasers because I haven't seen --

MR. EVANS: We are going to look at that a little bit more tomorrow in Chapter 9. We've got some diagrams.

CHAIRMAN STETKAR: Good.

MR. EVANS: They're not real detailed but --

CHAIRMAN STETKAR: Let me telegraph something for tomorrow.

MR. EVANS: Okay.

CHAIRMAN STETKAR: Because it's something that I came across, in something that I can't quickly find here. Section 9.2.5.4 of the SER now, not your CLFSAR, there's something that quotes a response to an RAI, and let me give you that number, so you might want to look it up.

It's RAI 6403?14.03.07-38 Part 3. If you're looking for letters, the letter is dated September 24th, 2012 and November 12th, 2012. Question was

regarding freezing in the ESW pipe tunnels and pipe chases.

And at least quoted in the SER, it says, ambient temperature in the ESWPT and ESWPC will not fall below freezing but will remain at or above ground temperature. The tunnel is not a closed area, so air can pass through it.

The tunnel openings are connected to heated areas in the reactor building, power source building and UHSRS, and only warm air passes into the tunnel area. Therefore water in the tunnel piping will not freeze.

That statement gave me the impression that if I was standing in the essential water service pump building let's say, I would see a pump going out and there would be an open space, or a pipe going out. There would be an open space around that pipe, and I can crawl through that open space and I will come to an open space going into the reactor building, and I will come to an open space going into the power source building.

In other words that those pipe tunnels, and pipe chases are not sealed compartments, because this says warm air can flow throughout that area and that's one of the reasons why the pipes won't freeze. And I was curious, if that's correct, then I'm not sure how the flooding analyses look at floods and potential

propagation of floods among those different locations.

Floods that either originate in the pipe tunnels or pipe chases themselves, or floods that originate in one location, and water flows through the pipe tunnels and pipe chases to another location. So if everything is closed and isolated, I understand.

But if indeed it's open the way, at least I got the impression from that question about nonfreezing of the pipes, that lead to a question about design for the flooding anyway.

MR. EVANS: Yes, I think I can understand your question, the --

CHAIRMAN STETKAR: We know we want to address it tomorrow --

MR. EVANS: We can do that or --

CHAIRMAN STETKAR: -- so if you have the drawings, let's do it tomorrow if you've got the drawing.

MR. EVANS: Okay.

CHAIRMAN STETKAR: But I just wanted to alert you, because it is a related type issue. So if you have drawings better, than show the configurations and whether the penetrations are sealed or open, we can address it tomorrow.

MR. EVANS: Yes, we can.

CHAIRMAN STETKAR: I won't forget.

MR. EVANS: We can be sure of that.

CHAIRMAN STETKAR: I have it written down, so I can't forget. But I did want to alert you to it because it was something I wasn't quite sure about, so.

MR. EVANS: We'll pull out, there are a couple drawings that are actually in Chapter 3, and actually the next slide kind of gets into that. There is an Appendix 3K, and there's a couple of figures, well this figure 3K-201, a couple of sheets there that show the, I don't have them, those figures in the slide, but they're in the --

CHAIRMAN STETKAR: Yes, I mean I looked at those and for some reason I wasn't, I still couldn't quite get the configuration in my mind.

MR. EVANS: They don't completely tell the story of, which you're talking about there, yes.

CHAIRMAN STETKAR: Okay.

MR. EVANS: So we've kind of already covered this slide, I think, so move onto Section 3.5.

MR. CONLY: Okay, 3.5, missile protection. Internally generated missiles from the pump house, vent fans, pumps, and cooling tower fans are not considered credible due to casings, housings, shrouds, and steel and concrete structures.

Comanche Peak Units 1 and 2 are outside the

low trajectory turbine missile strike zone, of Units 3 and 4, and vice-versa. Unit 3 and Unit 4 turbine generators are in an unfavorable orientation, as defined by SRP Section 3.5.1 with respect to safety-related structures systems and components of the adjacent US-APWR unit.

However, the inspection intervals keep P1 less than E to the minus five per year. Therefore, P4, the acceptable risk of turbine missiles, is maintained less than E to the minus seven per year.

CHAIRMAN STETKAR: Can I stop you before you go to the next slide, because I had two questions related to this slide. The first is, I have yet to understand what the words, not credible, means. So I'm not I don't understand that. I do understand, big concrete walls, I understand that sort of stuff in terms of confining missiles once they're ejected.

One of the questions that I did have, and this, I don't want to address the notion of not credible versus once in ten million years, because by saying not credible you're saying it's by definition, much, much less than once in ten million years.

The ESW pump rooms have the pumps in them obviously. They have fans in them, which have shrouds and all that kind of stuff. One of the questions that

I had, is are the cables for the UHS transfer pump, run through the ESW pump room? Here's the concern.

If I get a missile bouncing around the inside of the ESW pump room, can that missile damage the cables for the transfer pump for that division? Because I don't care whether the missile damages the ESW pump because that's clearly only one division. However, the transfer pump functionality for that division could be important.

MR. EVANS: I think what you're probably getting at is that the, can a given pump house, the ESW pump is one train, and the transfer pump is a different train.

CHAIRMAN STETKAR: And I'm not so much concerned, it the concern is, that suppose I do have a missile generated in that room, from some source. If that missile disables the ESW pump, that's fine. That takes that division, let's call it division A, train A out of ESW.

If I have another division drained for maintenance, which I can, I'm now left with two ESW pumps and the transfer pump from what, let's call it the division A basin, to make up for my 30 day supply for the two remaining divisions that I have operable.

So for example in particular, let's

generate the missile in train A, let's say that train B basin is drained for maintenance, which it could be during power operations. That leaves me train C and D operating and operable.

But I now have less than 30 day's worth of inventory available in trains C and D, because if that missile in train A disables my transfer pump, or basin A, I'm now left with less than 30 days water inventory, follow me?

MR. EVANS: Yes.

CHAIRMAN STETKAR: In that example, so the question is, are the cables for that transfer pump run through or vulnerable to damage from a missile in the ESW pump room itself or are they run outside of that room?

MR. EVANS: No, they are separate. The cables from one train do not go in the compartment for the other train.

CHAIRMAN STETKAR: Okay, they do not, okay, because there was some discussion, again in Chapter 9 that wasn't quite sure, it almost sounded like the cable, the cables were in the room, but separate.

You know, when I looked at fire analysis, it sounded like they were separated by distance, but still in that room. But if they're not run through that room, that alleviates my concern.

MR. EVANS: There have been some design changes over the evolution of the, and you referred to the RAIs and all that so I'm sure you're familiar with those. But in, so in earlier revisions, I think we might have had that situation.

The fact where the pumps may even have been in the same room, but the current design has a wall between the ESW pump and the transfer pump. And they're separate.

CHAIRMAN STETKAR: I saw the wall, but again, I'm concerned about that the cables would also only come up into that --

MR. EVANS: There's only cables from the ESW pump in that room, and they're only cables for the transfer pump, and associated equipment, in the train I should say.

CHAIRMAN STETKAR: Thank you, that answers that question. Let me just write notes. And the other question I had related to this slide, is that you heard the discussion about confirming that the frequency of turbine missiles is less than ten to the minus five for a year. And all of that questions.

Because you have unfavorably oriented turbines, regarding inter-unit configuration between Units 3 and 4, the conclusion is that last sub-bullet,

that the risk is acceptable because P4, which is the product of P1 and P2 and P3, is less than ten to the minus seven per year.

And my question is, how do we have assurance that the product of P2 and P3, P2 being the conditional probability that the missile strike, strikes a critical component, and P3 being the conditional probability that given a missile strike, damage occurs. How do we have assurance that, that probability is less than ten to the minus two?

Have you done an evaluation that looks at target areas for both low trajectory and high trajectory missiles to give yourself assurance that indeed that's the case?

MR. CONLY: Let me yield to MNES, Ryan.

(Pause)

MR. MINAMI: This is Minami speaking and I understand that, that the combination of P2 and P3 is described in the SRP at 3.5.1.3. And for favorably the strike can damage, should be one times ten to the minus three. And for unfavorably one times ten to the minus two, right? And if my understanding is correct, we did not need to verify the product of P2 and P3.

CHAIRMAN STETKAR: Thank you, I'll ask the staff and why they know that's less than 10 to the minus

two.

MR. CONLY: And by controlling P1, through increased inspection frequency, we drive that to where P2 and P3 can remain as stated in SRP 3.5.1 and yet P4 can be maintained less than E to the minus seven.

CHAIRMAN STETKAR: The implication of course is that even if I had all of my safety-related equipment immediately adjacent to every missile that flies out of that turbine, by some inference, the frequency of damage is less than one in a hundred. And to me that doesn't necessarily fall as a direct conclusion. So you're basic, what I hear you saying is you're basically relying on the numbers in the SRP, without doing any specific analysis.

MR. CONLY: That's my understanding, yes.

CHAIRMAN STETKAR: So, I'll defer that question to the staff, so that I understand why they have confidence then, that the product is less than ten to the minus two. Okay. Let's see if, I really have a closed session scheduled at 4 O'clock. Let's see if we, no we can't.

Yes. So in the interest of not stopping you in mid-sentence on slide 13, let's temporarily recess from this presentation, pick it up on slide 13, when we reconvene and Ryan, are you folks ready?

MR. SPRENGEL: Yes.

CHAIRMAN STETKAR: Okay, what we will do now is, we will close the meeting, so that we can discuss the questions regarding the steam generators. Girija, could you do me a favor and go find Sam, because he did want to sit in on this session. Harold does not, but Sam did. I alerted him to the fact that this would occur before, but I don't want him to feel to left out.

In the meantime, I'd asked both the staff and MHI to confirm that there is no one in the room who should not be here. So we're okay. And we are then in closed session. And I'll wait until Sam comes.

Make sure that we have the bridge line closed, if, it's open.

(Off microphone discussion)

CHAIRMAN STETKAR: The phone is fine, it's we have separate lines to the bridge line, I think. Not today?

(Off microphone discussion)

MEMBER ARMIJO: Some protection.

CHAIRMAN STETKAR: Do you have the bridge line open so that we can hear folks speaking or not? Okay.

MR. SPRENGEL: Then we should go.

CHAIRMAN STETKAR: No. But you do want

them to hear?

MR. SPRENGEL: Yes. Okay.

CHAIRMAN STETKAR: That's the best we can do then. With that, we are in closed session and let's address at least the questions that were raised this morning.

MR. SPRENGEL: I guess I'd like to give a little bit of an overview first, and then we can see how far to go from there. But basically because of ongoing litigation, we will have no further discussion on SONGS or any discussions related to SONGS specifically or testing for SONGS or anything.

Now we do have a RAI response that is in process and under discussion with the staff. We've given the response, and had some discussion, and given a followup response. And we're continuing discussion with the staff on that.

And actually there is more than one RAI response, but the most recent one right now is looking at making a, basically a design commitment for the in-plane fluid elastic instability ratio below .75. And then discussion with the staff most recently is focused on not only making that commitment, but giving the design information of how that evaluation is going to be done.

And that's our most recent step with the

staff, of giving some information on how that future design would happen. And it is future work. All these evaluations are reliant on the detailed design information that we don't have right now.

So we're making commitments on evaluations in the future, and we're giving a methodology. And just so you know, it's RAI number 1031-7031. So again, still draft, we're working on it right now.

But the second part of that goes into the methodology, and it is to basically apply the ASME methodology for out-of-plane vibration, or in-plane as well. And it goes through some details of adjustments and some of the coefficients and values that would be adjusted for the evaluation in the future of our getting -- so that's all in the second part of that RAI response going through the future methodology.

The first part gets into a question that came up earlier in terms of comparisons. And the staff had asked and we've expanded the comparisons, that right now include SONGS and also Fort Calhoun, and also Arkansas Nuclear 1, Unit 2.

And that was really for a middle ground in terms of size. So we've got SONGS is huge, Fort Calhoun was smaller, we were kind of in the middle but on the smaller side, now we Arkansas Nuclear 1 kind of filling

that gap in between. And we rule out different variables as, there's an RAI word, design features is what we call it.

So it gives a perspective of where the different plants fit, and we go through a discussion, not so much in this RAI, but in a previous RAI response, RAI 930, where we looked at kind of some of the root causes, and those are documented of course in the SONGS information, and those are publically available.

But particularly focusing on steam quality, and we looked at comparisons for SONGS versus US-APWR, and it goes through, you know, how much let's see, we have a lower steam quality, and the kind of the run down of effects as a result of that.

So this RAI is newer and the staff has asked us to kind of expand our comparison beyond just SONGS, and that's when we've added the other plants. So it gives some comparisons for sizes and just different features of the different plants.

So that's in this response, and that's what we're still working on, on finalizing. But ultimately it comes down to a commitment for the future evaluation for evaluating the in-plane FEI ratio. And maintaining it, now below, of course above one is the real problem, so we're adding a margin even below that and setting the

value at .75.

MEMBER ARMIJO: Okay. I had just a couple questions. The factor of .75, is that a well-established analytical basis to assure that you do not get this in-plane versus the out-of-plane?

MR. SPRENGEL: That was --

MEMBER ARMIJO: My understanding that past practice was that if you had met the out-of-plane requirements, you were sure to meet the in-plane requirement.

MR. SPRENGEL: That was in previous understandings, I would agree.

MEMBER ARMIJO: Yes, but kind of a folk lore antidotal, or whatever it was, but in fact it turned out not to be the case, and so maybe it's a good question to Pete, who's more familiar with the code. Of whether this .75 is kind of a well-established conservative thing, or whether you established this by virtue of testing? Because I know you, there was some testing done.

MR. SPRENGEL: Let me, okay so the .75 is actually a conservativism, and it's adding margin, so you're right in terms of previously looking at out-of-plane vibration, and you said, okay we're okay with that, we were wrong.

MEMBER ARMIJO: But we were wrong.

MR. SPRENGEL: Now we've accepted no, that's not good enough, let's also look at the in-plane vibration and the value we've had. So that's the new key part. And the .75 is not actually the real number, it's a ratio of 1.0, that's the problem. So we're adding this margin below that and saying, we're not going to deal with .99 and then you know, one up there. We're going to come down and keep our ratio below .75 as an additional margin.

MEMBER RICCARDELLA: What's the standard practice for the out-of-plane, as used?

MR. SPRENGEL: I think it's also .75, so that the key part --

MEMBER RICCARDELLA: That's standard practice all along?

MR. SPRENGEL: Right. The key part is not the .75, it's the evaluation being done at all.

MEMBER RICCARDELLA: Yes.

MR. SPRENGEL: And I will say that we've given a methodology, we've given approach right now, our intention is to stay engaged with the industry and I think things will continue to evolve. And we'll continue to adjust as any different evaluations are made available.

But we wanted to get something to the staff right now and make these in a way, commitments, that we're going to do this evaluation. We're going to do it. This

is all we have as an industry right now, it's the best we have. And as we go forward, we'll stay engaged and contribute as we can to improve any methodology.

Because I think one thing we're left with right now is the methodology is arguably over conservative. Because we just have to go with what we have. And so we're okay, we're documenting that, we're giving that to the staff, working out some details back and forth, but ultimately this is going to be an item that continues to develop over time.

And we'll stay engaged. I mean of course we have our stuff to deal with now, and we'll do that. But this will be an item just for the industry as a whole to continue to learn and develop knowledge on.

MEMBER ARMIJO: Yes, I keep coming back to the question, is you know, is the conservatism in the methodology that you're including a .75, but first of all start with the analysis, and then you apply a conservative factor, is that supported by testing, test data? Or is that just calculation?

MR. SPRENGEL: I cannot speak to the ASME methodology that being created. I think that's the root of the question and --

MEMBER ARMIJO: But it's not, so let me get one last try. It's not supported by Mitsubishi test

data? It comes from, if there was test data that was used to let's say, validate the ASME methodology?

MR. SPRENGEL: No, we're not --

MEMBER ARMIJO: That's up to, you have not done --

MR. SPRENGEL: US-APWR is not doing any testing to validate the ASME code methodology.

MEMBER RICCARDELLA: You know the ASME code has some, is based on some testing, but it's on very simple tube bundles straight tubes, and things like that, and it specifically says in the appendix that if you're going to use more complex, or different designs, you should do testing.

That's stated at, I forget the appendix number, I haven't, it's been a while since I looked at it. But I would guess that, you know, if your design, you know, we've been designing and building steam generators for a long time without having these problems, and so as long as the design isn't too much of a departure from existing designs, that would count as testing, to some degree.

MEMBER ARMIJO: Sure. Operating experience would count as testing, some experience.

MEMBER RICCARDELLA: I think you said you were going to make these comparisons, but these steam

generators are much smaller than SONGS, is that because it's four instead of two?

MR. SPRENGEL: Yes. The number of steam generators available in the different plants is a key part of it. Yes. Just reading off our table here, we have SONGS, I mean the key, internal power, yes. This is like a thermal output first steam generator, which is one of the key factors.

On SONGS that's 1729 US-APWR 1116. So I mean SONGS, just the reality of it is, you know, fewer steam generators, they're huge and they pump a lot of energy into them. And that's some of the detail that we go through between these two different RAI responses, highlighting that they are different. They are definitely different.

MEMBER ARMIJO: Okay. Thanks, that answers my question.

CHAIRMAN STETKAR: Yes, it sounds like, you know, in terms of our involvement, we'll just, whatever the schedule plays out to be, we'll just follow it as the RAI responses play out, and staff completes their evaluation of those responses. Both of you satisfied?

MEMBER ARMIJO: Yes, you know, I've asked the question and he's answered the question. You know, ultimately the staff will have to, I may ask it again.

CHAIRMAN STETKAR: Yes, sure. No, that's, but it's does in terms of --

MEMBER RICCARDELLA: Staff has asked the question --

(Crosstalk)

CHAIRMAN STETKAR: Yes, they have asked the question, so it's on the table. It needs to get resolved in terms of the staff. And we get a shot at it, you know.

MEMBER ARMIJO: In the future.

CHAIRMAN STETKAR: In the future when it is indeed eventually resolved, so. Anything else?

CHAIRMAN STETKAR: No.

CHAIRMAN STETKAR: Okay. Thank you, thanks Ron, that was really helpful. Then what we'll do is we'll reopen the session and ask Luminant to come back up and finish their presentations.

What I'd like to do today, if it's feasible, is actually finish this Luminant's presentation on Chapter 3, and the Staff's review on Chapter 3, so that tomorrow we can pick up Chapter 9 on Luminant in the morning. That's easier on everybody I think.

MR. SHUKLA: Yes, we already have slides for staff.

CHAIRMAN STETKAR: Yes. So thank you. And we are back in open session.

MR. CONLY: 3.5 missile protection continued. The probability of an aircraft crashing into the plant was estimated in accordance with SRP 3.5.1.6, because the plant is within five miles of the nearest edge of military training route, Victor, Romeo, -158.

The number of aircraft operations on VR-158 is less than the number of operations per year that would increase the crash probability beyond E to the minus seven. Therefore neither an air crash nor an air transportation accident is required to be considered as part of the design basis in accordance with the SRP.

CHAIRMAN STETKAR: Okay, I did some back of the envelop calculations and I won't bore you with those. My basic question is that in the FSAR, as far as military aircraft crash frequencies that are used in the analysis.

The assertion was made that the commercial aircraft crash frequency of four E to the minus ten, crash per flight mile, provides a conservative approach for determining the probability of in-route crashes on military airways. That's a quote from the FSAR.

I, for other activities, compiled a reasonable amount of aircraft crash data, and indeed a frequency of four E to the minus ten per flight mile, for what are typically characterized as in-transit flights of commercial aircraft operating under the rules of 14

CFR 121, which basically means the kind of aircraft that we normally fly around on, the commercial passenger aircraft that are subject to rather stringent rules.

That four E to the minus ten frequency, is reasonably comparable to actual experience. If you go into FAA and NTSB data bases over quite a long period of time.

It is crash frequencies for small aircraft, for aircraft that we generally consider to be commuter aircraft or normally just call it commuter aircraft, are considerably higher than that. Also and this is data you can't find from FAA, because they don't compile the data, crash frequencies for military aircraft tend to be quite a bit higher than that.

Particular this data available for, unfortunately the data that's available for military usually are not recorded in terms of crash frequency per flight mile. That's a problem. At least not in the U.S. In other countries they are, but I can't share that information.

MEMBER RICCARDELLA: The four E to the minus ten is for flight miles?

CHAIRMAN STETKAR: Four E to the minus ten is per flight mile in-transit. So we're talking about cruising aircraft crashes. We're not talking about

climb, or decent, or year airport operations, those crash frequencies are a lot higher.

But this is for just, you know, your basic cruise operation, climb over the desert, looking up, down, observing the scenery, falling out of the sky from 35,000 feet or whatever.

Military aircraft crash frequencies tend to be higher than that. At least from the data I've seen. So my fundamental question is, what's the justification for your assertion that the military aircraft crash frequencies are either comparable to or bounded by this four E to the minus ten crash frequency per year, per flight mile value? It's derived from commercial experience.

That's sort of the basic question. I have other questions about methodology and things like that, number of flights in air corridors, and I'm going to ask the staff about the number of flights. But I understand the methods that you've used. And I understand at least the data that are tabulated for the number of flights in that particular military corridor.

But I guess for you, because that number is quoted in the FSAR, I'd like to understand why it's either accurate or conservative for military aircraft crash frequencies? Considering the range of military

aircraft and that you typically find military aircraft, depending on, in the data bases I've seen for the U.S., they're typically not very specific. They tend to sub divide them into quote, unquote, "large military aircraft", which are things like bombers and cargo aircraft, and they're, you know, tankers. And small military aircraft that are fighters, and attack aircraft and training aircraft.

And the crash frequencies in particular for the small aircraft are about an order of magnitude higher than the larger aircraft. So I don't know what kind of aircraft used this particular air corridor, because I don't know, you know, I don't know what people do there.

It's not a, I pulled it up, it's certainly not a training area, which is good news for you. But it's a point to point corridor, and I don't know what kind of aircraft are using it, so I guess I'd like a little bit of feedback on the justification for that frequency.

MR. CONLY: We understand your question. The analysis was done in accordance with SRP 3.5.1.6, and we will have to get back with you on the details of that analysis.

CHAIRMAN STETKAR: Okay. Thank you.

MR. CONLY: Are there any other questions on 3.5 missile protection? Then let me ask Todd to take

3.6.

MR. EVANS: All right. Section 3.6 addresses protection against effects of postulated rupture of piping. There are no site-specific high energy piping within the essential service water pipe tunnels or ultimate heat sink related structures.

The essential service water pipe protection system contains site-specific moderate energy piping. Pipe break hazard analysis for site-specific moderate energy piping will be performed per the guidelines given in the DCD.

And will identify any postulated crack locations, require protective measures, and environmental and flooded impacts to ensure that safety-related functions are not affected. This pipe break hazard analysis is subject to license condition 3-1 which Don, showed us earlier.

And we will have procedures that will address plant operating and maintenance requirements to prevent water hammer. And that's all for Section 3.6. Any questions?

CHAIRMAN STETKAR: No, because you only own the pipe chases and the pipe tunnels. So, okay. Go on.

MR. EVANS: Okay. Move on to Section 3.9 on mechanical systems and components. The only

site-specific active equipment is the ultimate heat sink transfer system pumps and valves. The IST Program incorporates the DCD described program as expanded upon in the FSAR using ASME Section 11, ASME OM Code, Tech Specs, and good engineering practice.

And it is also the subject, and is also in conformance with NUREG-1482 guidelines for in-service testing. And also note that the IST program is the subject of license condition 3-3, which we covered earlier as well.

We do have one SER open item on Section 3.9, it's 3.9.6-21. It addresses the providing of additional details of the requirements for functional design qualification and IST programs as is typically given in specifications.

This is a subject that was mentioned and covered in the DCD presentation, so it's the same story here. That typical specifications are being prepared and will be available soon for staff review. Those are the same work, body of work that was discussed in the DCD presentation.

MR. WOODLAN: Okay, next we're moving to 3.10, and this is Don Woodlan. I'll cover 3.10, seismic and dynamic qualification of mechanical and electrical equipment. 3.10 and 3.11 kind of go together to cover

the equipment qualification and I'll cover both of those, starting with the seismic first.

We do use pretty standard OBE as one third of the SSE for Comanche Peak. The EQ that's the environmental qualification program, is described in MUAP-08015. Now MUAP covers more than just environmental qualification, it in fact covers just about the entire equipment qualification, including how we handle mechanical equipment.

The site-specific implementation requires the turnover of all the EQ program products to Luminant, and that's part of giving the operational program for the maintenance and environmental qualification.

Now to 3.11, the actual environmental qualification program. It will be established and implemented prior to fuel load. This is the operational side of the EQ program. Equipment test results are maintained as auditable project records and those will be turned over and maintained by Luminant as part of the program.

Site-specific equipment, the site equipment of concern are those that are subject to loss of ventilation and still required to remain functional. And those are addressed and identified in the FSAR.

In looking at Appendix 3D, is where we

identify the equipment, it in seismic Category 1, and these will require, be required to be qualified and addressed as part of the equipment qualification program.

We do have a, let me make sure, yes, two SER open items related in this area. The first one has to do with the environmental qualifications, specifications for mechanical equipment. And this issue's tied to the same issue we touched on several times, design and procurement specifications.

And one of the aspects that will be looked at is, do they adequately address the environmental qualification requirements?

The second open item is to provide a full description of the operational environmental qualification program. That is included in MUAP-08015. It has been provided to the staff and it is under review by them. And that takes us to 3.12. Any questions on 3.10 or 3.11? 3.12, piping design review. Todd.

MR. EVANS: All right. Regarding the piping design review for Section 3.12. Site-specific response spectra is used for analysis of yard piping that's not included in the standard plant design in the DCD. ASME 3, Class 2 or 3 piping is not exposed to wind, or tornado loading for the site-specific piping.

Exposed non-ASME piping is evaluated for effects.

For site-specific ground motion response spectra, there are no exceedances of high frequency certified seismic design response spectra. Therefore, screening of piping for high frequency sensitivity is not required.

Section 3.13 covers threaded fasteners, basically what's included in the FSAR is our commitments for the programs associated with this. Threaded fastener Pre-Service Inspection Program will be implemented after start of construction.

And the In-service Inspection Program will be implemented as part of Operational Program. And that concludes our Chapter 3 presentation for the COLA.

CHAIRMAN STETKAR: Do members have any more questions for Luminant? Good. Thank you. Wasn't too painful.

(Off microphone discussion)

CHAIRMAN STETKAR: You notice Don is getting really good about understanding which sections he ought present.

MR. EVANS: He is good isn't he?

CHAIRMAN STETKAR: Isn't he good.

MR. EVANS: He knows which ones to choose and I see he disappeared.

CHAIRMAN STETKAR: So we'll have the staff come up and give their presentation on the review, Chapter 3.

(Off microphone discussion)

MR. GALVIN: Did you turn it on? Girija, I think I need help.

MALE PARTICIPATE: And you'll need the password.

MALE PARTICIPATE: I guess we could work with the handouts.

(Off microphone discussion)

CHAIRMAN STETKAR: Dang that's a good password.

(Laughter)

(Off microphone discussion)

MR. GALVIN: Yes, but I memorized it. Whole bunch of periods.

CHAIRMAN STETKAR: Okay, we're ready.

MR. GALVIN: Again, we are briefing on the US-APWR, on looking at Comanche Peak, Chapter 3, SE. We've identified the reviewers. We have a little thinner staff and I think we have the ones, at least the questions you've identified so far, they're still here. So if you have any questions.

CHAIRMAN STETKAR: All right.

MR. GALVIN: We have three open items as was discussed before. The first two involve conducting an audit of the design and procurement specifications for the in-service testing. Well, address implementation of the functional design, qualification, and in-service testing programs and also for environmental qualification of the mechanical equipment.

And the other open item is the staff cannot complete its evaluation of Luminant's description of its environmental qualification operational program, because the review of the US-APWR Equipment Environmental Qualification Program is incomplete.

We still have a fair number of open items or issues as you heard this morning, so we're, it's just not practical to complete the review until we make more progress for the DCD. That's all we have. Those are all the open items, so if you have some questions?

CHAIRMAN STETKAR: Dennis, do you have your aircraft crash reviewer here?

MR. GALVIN: Right here.

CHAIRMAN STETKAR: Excellent.

MR. GALVIN: Raul is here. Time to run away.

CHAIRMAN STETKAR: Two questions that I had. One is, and I don't know whether this is a typo

or not. In Section 3.5.1.6 of the COL FSAR Revision 3, there is a statement that says, in order to maintain PFA, which is the aircraft crash frequency, less than the order of ten to the minus seven for both Units 3 and 4, the above equation is rearranged to solve for N, which is the number of flights, the flight operations per year, using values CA and W, is determined below.

And the conclusion is that as long as the number of flights in that air traffic control, air traffic corridor VR-158 is less than 19,300 flights per year, given that aircraft crash frequency that we discussed earlier, four E to the minus ten, crash per year, per flight mile, everything is okay.

And indeed I confirmed, given the configurations, and distances, and areas, and things that, that 19,300 value is consistent with the information in the FSAR.

In Section 3.5.1.6.4 of the safety evaluation report, is a statement that says, based on the guidance provided in SRP Section 3.5.1.6, the applicant estimated the number of flight operations is required to be less than 35,300 per year to meet the acceptable total probability of an aircraft accident of less than ten to the minus seven per year.

So now I'm curious, why in one place it says

19,300 and in the SER it says 35,300?

MR. AHMED: In the version of FSAR, 19,300 was calculated. But later on, they have changed the way they calculated the area, to be 0.0909 and also they have many other, in the earlier calculation with respect to the actual distance that had to be used.

Therefore we corrected their calculation and submitted the revised calculation to the staff. Therefore based upon that calculation if, we confirmed, the value would be 35,300.

CHAIRMAN STETKAR: Okay now --

MR. AHMED: Yes.

CHAIRMAN STETKAR: Okay go on.

MR. AHMED: Yes. And then to go from the -- whether it needs, I mean, they have calibrated and confirmed that they're meeting 35,300 in order to meet the probability criteria of one ten to the four minus seven. And the staff inquired the real --

CHAIRMAN STETKAR: Let me stop you there, because I have another question about that, but let me stop you at that point, because now I'm even more confused than I was before.

It's a pretty simple calculation, I used the information in the FSAR of .0909, a square mile per unit. I took, multiplied that by two, and I rounded it to .18.

I used the width of the aircraft traffic control corridor and the distance from the edge of the aircraft control corridor, as per the guidance in the SRP.

And I used a crash frequency of four times ten to minus ten per flight mile year. And indeed I came up with a value that was in round numbers 19,300, not 35,300. So I'm not sure all of the numbers that you quoted are the numbers that I used to come up with 19,300.

MR. AHMED: The area, original, the area was 0.07. Okay, that did not, we didn't change it. I did not degrade the criteria. But the other number is twice the width of the area, plus the distance.

CHAIRMAN STETKAR: That's right. That comes out to be 14 miles.

MR. AHMED: No, it will be, the width of the area is ten miles, times 20, two is 20, plus 7.6. 7.6 miles is the actual area from the, I think that is the distance.

CHAIRMAN STETKAR: 7.8, but I'll give you 7.6.

MR. AHMED: So if you take the width of the area is ten, and multiply by two, that would be 20 plus 7.6. 27.6 if you use that number, you'll get 35,300.

(Pause)

(Crosstalk)

MR. WOODLAN: Yes, this is Don Woodlan again. There has been some changes. And these will appear in Revision 4, next week I hope. The, you're correct, John, the number that we've come up with now and it includes some minor adjustments, but basically it was we incorrectly addressed the area by a factor of two.

Really the version that you saw and the current number that's going to appear in Revision 4 is 17,600 operations per year, which I think is aligned with what you came you up with, John.

CHAIRMAN STETKAR: That's, yes, I mean that's a lot closer, that's right, 17,600 and that's for, but that's half of the 35,300.

MR. WOODLAN: Correct.

MR. GALVIN: I think they told us 35,000, that's what made it in the SC, and I think it was a big, so maybe we must have --

(Crosstalk)

CHAIRMAN STETKAR: It may be this thing about hitting a single unit, versus hitting a both units.

MR. WOODLAN: I think that's exactly what's happened --

CHAIRMAN STETKAR: By a factor of two going on here, but okay, I understand that, okay. Got it. Let me just write a note here.

(Pause)

CHAIRMAN STETKAR: Just make sure that we get that straight because again I was using the total footprint, regardless of how you adjust the numbers, I was using the total footprint of the site versus the footprint of a single unit. And that's of course a factor of two in there.

Now, all right. Well, let me ask you the question because I cut you off, because I did have a question about your data. In the SER it says the staff obtained, latest five years, 2004 through 2008 of flight operations data within ten miles of the site from the federal aviation administration.

Using this data, the NRC staff determined that the maximum annual number of total military flight operations within ten miles of the site is 11,192 for the year 2006.

And my question is, in the FSAR, at least the version of the FSAR that I have, in Section 2.2.7.2 of the FSAR, it lists a number of flights per year of 300 to 400 for that particular military airway. And the other military airway is IR-139 and that lists 10 flights per year.

My question to the staff is, this 11,192 flights per year, which are characterized as military

operations from the FAA, is that the total number of flights, commercial plus military?

MR. AHMED: No.

CHAIRMAN STETKAR: Only military?

MR. AHMED: Okay, the FAA flight I get includes all flights from runway to destination, within five miles. So they give, where is the origin, where is the destination, date, and the flight, and designation of the aircraft.

CHAIRMAN STETKAR: Yes.

MR. AHMED: So there are prior designations they have.

CHAIRMAN STETKAR: That's what I have.

MR. AHMED: So did you get the, so first of all when the data comes, if it is not a designated military, I use all the aircraft flights. Using the ten miles to see very conservative estimation of the probability.

But in this case I have generated a program, I can thin out based upon the designation whether it is a military, so I screen only military.

CHAIRMAN STETKAR: So this is, this count of 11,192 --

MR. AHMED: All the aircrafts designated military by that letter, yes.

CHAIRMAN STETKAR: Okay.

MR. AHMED: Passing too within ten miles of the site.

CHAIRMAN STETKAR: Well that actual, then I'm really interested in the basis for that four E to the minus two, ten, four E to the minus ten crash frequency for military aircraft for in-flight operations because if indeed there is something on the order of 11,000 flights per year in that --

MEMBER SCHULTZ: 2006.

CHAIRMAN STETKAR: In 2006, but in that corridor, military flights.

MR. AHMED: I compiled the information by year, five years, and then --

CHAIRMAN STETKAR: And that was the highest of the five years?

MR. AHMED: That would be the highest of the five years.

CHAIRMAN STETKAR: Yes. And that's fine. But it's important for me, I don't care about the specific number or the year, it's on the order of something between ten thousandish versus a few hundredish. And that's a big difference.

MR. AHMED: Maybe a few hundred probably is pertaining to that, you know, one particular military

area. I do not know. I mean, it's a very difficult to designate that way.

CHAIRMAN STETKAR: Okay.

MR. AHMED: And to answer one more question about the probability, I looked at the data base compiled by Sandia National Laboratory for the various DOE sites. And again it is probably is by different aircrafts also. I looked that, to see a value how much it is, the probability is changed. So it is more or less comparable of that, as far you stated it could be an order of magnitude difference.

CHAIRMAN STETKAR: Yes. It's, did you look by the way at, there's a DOE standard. It's DOE standard, DOE STD.

MR. AHMED: Yes, that is --

CHAIRMAN STETKAR: 30142006, okay. That's the number that was used and it was compiled by Sandia, it was used for Yucca Mountain.

MR. AHMED: But, that was read, is also available.

CHAIRMAN STETKAR: Yes. I didn't have access to lab data, but I'm sure you guys do. And there they do differentiate between aircraft operations, take off and landings versus in flight, they give ranges, minimum average, maximum, for the two different, at least

in the summary information for the two different classifications.

And I've done some playing with that data, in terms of looking at different ways of estimating crash exposure areas, and backing out a crash rate per flight mile if you will. And they tend to be quite a bit higher than the commercial aircraft crash rates.

The large aircraft is more comparable. They're fairly close within a factor of three or four or five. But the small aircraft is a lot higher. They're like an order of magnitude higher.

So the concern is that if the air traffic density of 11,000 or so flights per year has a high number of small military aircraft, that it's not clear that the frequency, the crash frequency is on the order of ten to the minus seven.

MR. AHMED: But probably we have to keep in mind if it is a smaller aircraft, even though it is probably is higher, the impact hazard is much less.

CHAIRMAN STETKAR: But they didn't do that analysis, did they? They simply argued that the frequency of an aircraft of any size or any energy within the site, is less than ten to the minus seven.

A more detailed analysis might justify the fact that the risk is acceptably small, but they didn't

do the detailed analysis. And right now I'm kind of questioning, you know, whether or not the assertion that the crash frequency, regardless of potential damage, indeed is less than ten to the minus seven.

And as I said, if the, if your flight density data are at all in the right ball park, on the order of 10,000 or so flights per year, then it becomes even more important to understand whether or not the frequency that they're using for the in-flight crash frequency is indeed reasonable. Because you're starting to get really close actually. Really close.

MEMBER SCHULTZ: And there's a difference to be explained.

CHAIRMAN STETKAR: Yes, so.

MR. AHMED: And we thought we were answering one letter to ACRS long time, three years ago with the respect to viewing the details.

CHAIRMAN STETKAR: Yes, I knew, yes, that's why I didn't talk much about the methodology. We've discussed that before, so I was, stick to this methodology and at least discuss the numbers.

Thanks that helps a lot by the way. Any other members have any questions, further questions for the staff? If not, let me do a couple of administrative things here.

First of all, let me ask if there are any members of the public in the room who would like to make a statement or comment? And I don't know, I know we have a bridge line, I don't know if there are any members of the public on the bridge line?

MEMBER RICCARDELLA: No.

CHAIRMAN STETKAR: There are not.

MEMBER RICCARDELLA: Just whatever you put on the tape.

CHAIRMAN STETKAR: Okay. So I don't need to open the bridge line to see if there are any public comments. I'd like to thank the staff for your presentation on Chapter 3. And as we usually do at the end of the subcommittee meetings, I'd like to go around the table, and see if any of the members have any closing comments or areas that they'd like to bring up in summary, because this is our shot at Chapter 3. Tomorrow we'll talk about Chapter 9. Pete?

MEMBER RICCARDELLA: No.

CHAIRMAN STETKAR: Harold?

MEMBER RAY: I did listen, and no I don't.

CHAIRMAN STETKAR: Thank you much, Steve?

MEMBER SCHULTZ: Nothing further.

CHAIRMAN STETKAR: Charlie?

MEMBER BROWN: Nothing further.

CHAIRMAN STETKAR: You notice I leave the best for last. Sam?

MEMBER ARMIJO: Nothing more.

CHAIRMAN STETKAR: Thank you, and I don't have anything more, so miraculously we're finished and we'll talk about Chapter 9 of the COL tomorrow. And we are adjourned.

(Whereupon, the meeting in the above-entitled matter was concluded at 4:53 p.m.)

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NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

+ + + + +

US-APWR SUBCOMMITTEE

+ + + + +

THURSDAY

NOVEMBER 21, 2013

+ + + + +

ROCKVILLE, MARYLAND

+ + + + +

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

COMMITTEE MEMBERS:

JOHN W. STETKAR, Subcommittee Chairman

CHARLES H. BROWN, JR. Member

HAROLD B. RAY, Member

PETER C. RICCARDELLA, Member

STEPHEN P. SCHULTZ, Member

DESIGNATED FEDERAL OFFICIAL:

GIRIJA S. SHUKLA

ALSO PRESENT:

EDWIN M. HACKETT, Executive Director, ACRS
SARDAR AHMED, NRO
MARK BIERY, MNES
PERRY BUCKBERG, NRO
THERESA CLARK, NRO
JOHN CONLY, Luminant
ANTONIO DIAS, NRO
JENNIFER DIXON-HERRITY, NRO
TODD EVANS, Luminant
DENNIS GALVIN, NRO
RAUL HERNANDEZ, NRO
THOMAS HICKS, MNES
JOHN HONCHARIK, NRO
MASASHI ITO, MNES
PAUL KALLAN, NRO
PETER KANG, NRR
RON LAVERA, NRO
RENEE LI, NRO
ATSUSHI MATSUMOTO, MHI
YOSHIHIRO MINAMI, MHI
STEPHEN MONARQUE, NRO

MASATOSHI NAGAI, MNES
HIROKI NISHIO, MHI
YUKIHIRO NISHIZAWA, MHI
RYAN NOLAN, NRO
HIKARU OGASAWARA, MHI
RICHARD SAMPLES, MNES
THOMAS SCARBROUGH, NRO
MOHAMED SHAMS, NRO
RYAN SPRENGEL, MNES
ANGELO STUBBS, NRO
YOSHIHIRO TAKAYAMA, NRO
LARRY WHEELER, NRO
DONALD WOODLAN, Luminant

*Present via telephone

T-A-B-L-E O-F C-O-N-T-E-N-T-S

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

(8:31 a.m.)

CHAIRMAN STETKAR: Okay, 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.

ACRS Members in attendance are Harold Ray, Steve Schultz and Charlie Brown. Mr. Girija Shukla is the designated Federal Official.

Today we're going to continue the discussions from yesterday and hear about Chapter 9 of the Comanche Luminant combined license application. I understand that we're going to have both opened and closed sessions today.

The way we'll organize it, I think Luminant will present the open session material first. We'll close the meeting and hear the closed session presentations and then reopen it for the staff's presentation.

I also understand that MHI has some responses to a few of our questions anyway, from yesterday, and that the staff had some additional information that they'd like to provide to support some of the discussions from yesterday. We will pick up those items after we finish the discussions about Chapter 9.

Again, I remind everyone to please turn of your cell phones so that we make sure that we hear you on the transcript. If you have something to say please come up to the microphone and announce your name and speak clearly and sufficiently so that we understand you and whatever those words say.

With that I will turn it over to Stephen Monarque to say, if you have anything to say as an introduction and he's shaking his head.

MR. MONARQUE: I just wanted to say, thank you for having us today. And with that we can turn over the mic.

CHAIRMAN STETKAR: Welcome, it's good to be had. I'll turn it over now to Luminant, Don Woodlan.

MR. WOODLAN: Thank you and good morning everyone. We are covering Chapter 9 this morning, auxiliary systems. Next slide, John.

CHAIRMAN STETKAR: And just for the record we've been joined by Pete Riccardella. Just to make sure that you're, acknowledge that you're here.

MR. WOODLAN: First slide here is the agenda. Again, it looks a lot like the agenda we normally use.

We'll start out with an introduction, address the ACRS open items, the ACRS license condition

and then the site specific aspects of our Chapter 9. And as you mentioned, there will be a closed portion there.

And we'll do that at the end unless we need to reference drawings during the discussion and then we can decide if we want --

CHAIRMAN STETKAR: Yes, we'll see.

MR. WOODLAN: -- go through it.

CHAIRMAN STETKAR: And as I said it, if some of our questions drift into areas where you think it's better to support the answers with that information, we'll just postpone it. We'll try to keep it all open and then transfer into close --

MR. WOODLAN: I think that will --

CHAIRMAN STETKAR: -- if that works. Okay.

MR. WOODLAN: All right, next slide, John. Here's the introduction. Again, very similar to the one we normally use.

We are continuing to use the IBR methodology. There are no departures from the US-APWR, DCD.

All COLA items from the DCD have been addressed in the FSAR. We do have one SER open item which we'll address. We do have a license condition which we will also address. And there are no contentions pending

before the ASLB.

Now for the site specific presentation I'm going to turn it over to Todd Evans.

MR. EVANS: All right, my name is Todd Evans with Luminant and I'm glad to be here today to discuss Chapter 9 of the COLA for Comanche Peak 3 and 4. This slide discusses the safety evaluation open item. We have one open item for Chapter 9.

The open item deals with standard design heat load calculations being adjusted which may impact some parameters in the COLA. Changes are expected as we go forward.

The heat loads for the ultimate heat sink for the service water are the load and coolant water heat exchangers and the essential chillers. So there will likely be some minor adjustments.

We do not expect those to impact the conclusions for the ultimate heat sink design but we want to make sure we get the capacities and the tables and everything correct ultimately in the COLA. So that's the reason for the open item.

Final parameters will be reflected at future updates to the DCD and then those will be reflected in the COLA.

CHAIRMAN STETKAR: Todd, are you going to,

I'm trying to page forward here and I'm not sure that I'm seeing it, I had a question about the heat load calculation. Is it the appropriate time to ask that question or do you have another slide --

MR. EVANS: I don't have any specific slides that cover the heat load calculation, so I'd say go ahead.

CHAIRMAN STETKAR: Okay, let me take a shot at it. If I read the FSAR there's analysis that was done, this is in Section 9.2.5.2.3 if you want the reference, the pointer for the FSAR section.

But there's an analysis that was done that used what is cited as the worse 30 day period wet bulb temperature based on data that were collected from, 30 years of data, 1977 through 2006. And I believe that that calculation was performed to evaluate the minimum heat removal from the UHS.

Because that's typically what's done with that type of calculation. Now I was curious, a lot of times I see calculations that also are designed to quantify the maximum evaporative heat loses so that essentially the maximum mass loss from the basin.

And those are typically done with a, it's not a minimum wet bulb temperature because that doesn't necessarily give you the maximum evaporated heat loss,

but it's a much different temperature profile where you maximize the difference between the dry bulb temperature and the wet bulb temperature rather than trying to minimizing the difference.

And I didn't see that type of calculation. I was wondering whether the, I'm not, I don't design cooling towers so I don't know how these actually work, but I only know from experience that I have seen two different sets of calculations performed in the past.

One to essentially minimize heat removal and the other to essentially maximize evaporative heat losses and the temperature. They're both based on a temperature profile that's derived from actually weather data, but in one case you try to minimize the difference between the wet bulb and dry bulb temperatures, the other case you try to maximize the difference over that period.

So I was curious whether the calculations that you've done, and I ask the staff too because I'm sure they have people who understand this much better than I do, whether we have assurance that you've calculated the maximum evaporative heat losses which actually then feed into the volumes of the basins and things like that?

MR. EVANS: I think we'll have to check on that unless maybe we can back with you. To my knowledge I've only seen the one calculation so I'm, but maybe I

haven't seen all the calculations.

CHAIRMAN STETKAR: Okay. We'll ask the staff, make sure that you've got your --

MR. WOODLAN: Well I know we did the calculation because we had to confirm that we had sufficient water for 30 days of cooling.

CHAIRMAN STETKAR: Right.

MR. WOODLAN: So I know we did that calculation.

CHAIRMAN STETKAR: Well this calculation, the section that I cited is all part of the discussion of that analysis.

MR. WOODLAN: Yes.

CHAIRMAN STETKAR: And yet as best as I can tell, the words that I read, I haven't see the details of the calculation, the words that I read seem to support the fact that you have adequate heat removal to satisfy 30 days under the most limited heat removal conditions. Most limiting atmospheric heat removal conditions.

MR. WOODLAN: Yes.

CHAIRMAN STETKAR: And it wasn't clear to me that that same calculation would support the maximum evaporative losses. Although both of those issues are discussed in the same section of the FSAR.

As I said, I don't design these things and

I have trouble boiling water on a hot plate, but I have for other cites seen two different calculations. And I thought I understand why, so.

MR. WOODLAN: We'll have to take away.

CHAIRMAN STETKAR: Okay, yes. And again, I'll ask the staff, perhaps you might have some of your quotes --

MR. MONARQUE: Yes, we'll be able to answer it.

CHAIRMAN STETKAR: Good, thanks.

MR. WOODLAN: What section was that by the way?

CHAIRMAN STETKAR: It's Section 9.2.5.2.3 on this slide.

MR. WOODLAN: Got it, thank you.

CHAIRMAN STETKAR: In the FSAR.

MR. WOODLAN: Okay, we get it.

CHAIRMAN STETKAR: The only reason I brought it up in the context of this, I read the SER and a lot of their questions tended to, I think, point to do you have sufficient capacity under the minimum heat removal conditions --

MR. WOODLAN: Right.

CHAIRMAN STETKAR: -- rather than this, the mass for the maximum evaporative heat losses. Okay,

thanks.

MR. EVANS: All right, for Chapter 9 we have one license condition discussed in the safety evaluation and that we've committed to. Basically it deals with the implementation of the fire protection program.

There are several different milestones for implementing different aspects of the fire protection program during the course of the project.

The first milestone is that storage buildings for byproduct and special nuclear materials. The program will be implemented before initial receipt of those materials.

And the second milestone deals with implementing the fire protection program in areas containing new fuel before receipt of the new fuel. And then finally the ultimate implementation of the full fire protection program features will be before initial fuel load.

Moving onto Section 9.1.5, the heavy load handling program. The aspect of this program or this section in the COLA, since we're mostly incorporated by references referring to the establishing of program prior to first fuel load, aspects of the heavy load handling program are detailed in Section 9.1.5 and the program will be established prior to first fuel load.

Section 9.2 and 9.5, these sections will be discussed in the closed session as we talked earlier, so let's put a place holder here to show that we are going to cover those sections.

So moving on then to 9.4, which covers ventilation systems. The capacities of the HVAC, heating ventilation air condition equipment, reflects site specific conditions.

The ultimate heat sink central service water pump house ventilation system maintains proper environmental conditions. So the COLA takes the site specific aspects for environmental temperatures and uses those to come up various capacities for HVAC equipment.

CHAIRMAN STETKAR: Todd, as usual, I have an off the wall question but it just struck me as curious. And the staff had a related question but a little bit different.

And this is, the terminology is somewhat obscure so I want to make sure we all understand what I'm talking about. There are HVAC equipment rooms that contain ventilation equipment, as best as I can tell, and that ventilation equipment supplies the main control room and the Class 1E electrical rooms.

So I'm not talking about the main control room itself or the Class 1E electrical rooms, I'm talking

about the rooms that has the HVAC equipment that supplies those locations.

And they have four trains of those rooms and because the COL Applicant had to supply the heating capacities, the heater capacities, they're induct heaters, the staff noted it and I started to followup and think about it, that according to the information in FSAR Table 9.4-201, that information indicates that there will be induct heaters in the ventilation supply to this ventilation equipment room.

So now I'm talking about the ventilation or the ventilation equipment room. But heaters are only needed for Trains B and C. That heating is not required for Trains A and D.

And I was curious because I couldn't think about what differences there would be among the four trains. If I look at the HVAC system layout in the P&IDs in the FSAR and the DCD, it all seems symmetric among the four trains.

And yet it's clearly said, in a response to one of their, they didn't ask why not have it in Trains A and D, they were questioning whether the heating capacity was sufficient in Trains B and C. And you responded in a letter that said, Applicant agreed to amend Table 9.4-201 to indicate that for the line item

of MCR class 1E, electrical HVAC equipment room, induct heater capacity that no heating by induct heaters is required for Trains A and D.

What, do you have any information about the asymmetries, why they're not required?

MR. EVANS: Not specifically, and maybe we can take that away and look at that, but my understanding is based on the location of them relative, there was not the need for the heaters.

CHAIRMAN STETKAR: Okay.

MR. EVANS: And the heaters in Train A and D are fairly small. They're only --

CHAIRMAN STETKAR: Yes, it's not a big issue it's just I'm always intrigued --

MR. EVANS: Yes.

CHAIRMAN STETKAR: -- in a plant that's otherwise, you know, very, very symmetric among the four trains, if I see sources of non-symmetry, I'm also intrigued about why.

MR. EVANS: If I remember correctly A and D, Trains A and D are on the outside so to speak --

CHAIRMAN STETKAR: It probably is.

MR. EVANS: -- and B and C are in the middle. So there could be something to do with how things communicate, but.

CHAIRMAN STETKAR: It's not a major point it's just something that, as I was reading through things was curious.

MR. EVANS: And what I was saying was, Train A and D, the capacity of those is 2.2 kilowatts which is, you know, very small heater. So it doesn't take a lot for the Train A and D whereas, so.

CHAIRMAN STETKAR: Okay.

MR. EVANS: But we can take that away.

CHAIRMAN STETKAR: Okay, yes it's a curiosity more than anything else. So don't make a big deal about it, please.

MR. EVANS: Okay. No, I noticed the same thing when we were going through this.

CHAIRMAN STETKAR: Well it was just notable also that the response to the staff's REI says that no heating is required for those rooms, you know, rather than saying, oh, yes, well it should be equal. Anyway, I'm sorry, continue.

MR. EVANS: Okay, next we'll cover Section 9.5.1, fire protection system. Basically again this deals with the program aspects of the fire protection system.

The fire protection program ensures that fire will not affect safe shutdown capabilities. We use

a defense and depth approach that includes fire detection, extinguishing systems and equipment, administrative controls and procedures and training personnel.

Our hazard analysis is performed and we will have a combustible control program as part of that as well.

CHAIRMAN STETKAR: Okay. I had a few questions about the fire protection program. Let me find my notes here.

Bear with me, I'm now completely disorganized. Let me ask you about the fire brigade first.

And it had some discussion about this I know before when we were talking about Chapter 13, so I in some sense, some of this information I'd like to confirm that the thought process is still the same and then I wanted to ask a little bit more focus questions specifically about the fire brigade.

According to the FSAR, the fire brigade is a five person fire brigade.

MR. EVANS: Correct.

CHAIRMAN STETKAR: And it's my understanding that you plan share that fire brigade between the two units, 3 and 4. Is that still the plan?

You basically have one site fire brigade that can handle fires at both units or is that evolving?

MR. EVANS: That I don't know.

CHAIRMAN STETKAR: Okay. Well some of that discussion back in, you probably remember --

MR. EVANS: Yes.

CHAIRMAN STETKAR: -- it was September of 2012 so I'm sure that it's right at the fore front of your memory. And at that time you said you hadn't really made final plans for the fire brigade but that at that time you were still planning to share the fire brigade between the two units.

So I wanted to followup, for my understanding, because some of my questions about staffing the fire brigade might be different depending on how, whether you have it organized with two fire brigades, one per unit versus a single fire brigade at the site level.

And let me get the more specific questions while you think about the first one. In the, and again I'll point you to the section of the FSAR just so that you have it for reference. It's FSAR Section 9.5.1.6.1.6. I love these things.

MR. EVANS: Yes.

CHAIRMAN STETKAR: The discussion talks

about the fire brigade but it also notes that during a fire event it says, an incident commander designated by plant management and trained in emergency services incident management systems, assumes the overall responsibility in the event of a fire emergency and provides advice and guidance to the shift manager. The incident commandeer is responsible for incident command activities and for making, following specific recommendations to the shift manager based on an assessment of the magnitude of the fire emergency from reports received from the prior team leader.

And some of those responsibilities include decisions regarding safe shutdown of the plant that's required, implementation of the emergency plant. And there's some other administrative duties.

Now I'm starting to get a little bit confused about the number of people and the responsibilities of the individuals on shift. In particular the shift manager, as I understand it, has the overall responsibility for operations of the plant and decisions regarding emergency planning, communications.

In other words that person is the lead responsible person for managing the plant. And those are fairly specific duties.

Each unit, as I understand it, will have a

shift manager and a unit supervisor. I think that --

MR. EVANS: Correct.

CHAIRMAN STETKAR: -- that we had some of that discussion during Chapter 13. So now if I think about a single unit, I have a shift manager and a unit supervisor, both of who are SROs.

And then you have a shift technical advisory, that according to our previous discussions, you plan to share between the two units.

So during a fire, now I have a shift manager who has a lot of responsibilities for overall operation of the unit and external communications and implementation of the emergency plant, I have the unit supervisor, whose a SRO who should be dedicated to minute by minute supervision of people manipulating equipment and following abnormal emergency operating procedures, fire response procedures and so forth, and now I have an incident commander who is making decisions about, should we shutdown the unit and making recommendations from, receiving information from the fire brigade or what's called the fire team, making recommendations to the shift manager regarding emergency responses.

Who fulfills all of these duties during a real event? I mean who do you satisfy the staffing for the shift manager, the incident commander, the unit

supervisor and the shift technical advisory those four responsibilities with the planned shift manning that you have and the responsibilities of the fire brigade that you've outlined?

We'll go through the fire brigade, the people who are actually out in the plant doing the firefighting next.

MR. EVANS: Well the --

CHAIRMAN STETKAR: I didn't quite, you know if there's an organization chart that points me to who does exactly what or I couldn't find one?

MR. EVANS: Yes, let me see if I could understand the, the shift manager, the unit supervisor, those are individual people.

CHAIRMAN STETKAR: Right.

MR. EVANS: The incident commander that's referred to here would be a different person from those. And as you correctly indicated, the incident commander does not have the decision making, he doesn't make the decisions as far as --

CHAIRMAN STETKAR: Right. As I understand it, the function is to collect information coming in from the field, process that information and make recommendations --

MR. EVANS: Make recommendations to the

shift manager --

CHAIRMAN STETKAR: -- to the shift manager, yes.

MR. EVANS: -- who would make the ultimate decisions.

CHAIRMAN STETKAR: Yes.

MR. EVANS: And yes, those positions would be, the incident commander would come from some of the on shift, other on shift personnel.

CHAIRMAN STETKAR: Well that's exactly my question. Which other on shift person?

MR. EVANS: Oh.

CHAIRMAN STETKAR: Because I'm starting to run short of numbers of bodies to do things like fight the fire, make decisions, do communication, you know, stand at the boards and actually manipulate control, supervise the manipulation of those controls and provide things like the shift technical advisor stand back oversight function from a plant safety perspective.

MR. EVANS: Well there are other on shift personnel. I don't have a list of --

CHAIRMAN STETKAR: Okay.

MR. EVANS: -- who those are or how many there are. But like the plant equipment operators are --

CHAIRMAN STETKAR: So this incident commander slot, that's what I didn't understand, it could be staffed by a non-licensed individual?

Because equipment operators don't have a license. And what concerned me a bit is one of the duties of this incident commander, at least as it's laid out in the FSAR, is to make recommendations regarding safe shutdown of the plant and implementation of the emergency plan.

MR. EVANS: Right.

CHAIRMAN STETKAR: Which is something that equipment operators usually don't understand a lot about.

MR. EVANS: But with the required training, I know I'm familiar with being involved in emergency planning, drills, I'm a member of emergency planning team for Units 1 and 2. I don't have an SOR license but as part of the duties in the TSC, we do make recommendations, we do the guideline for following the emergency plan guidelines, we're trained in them what the guideline are.

CHAIRMAN STETKAR: Okay.

MR. EVANS: We make recommendations to the emergency coordinators who are SRO type qualified. We don't make the decisions --

CHAIRMAN STETKAR: Yes, no, sure.

MR. EVANS: -- but we have enough training that we, you know, are able to make recommendations.

CHAIRMAN STETKAR: Okay, let me just make a couple of notes here. Okay. Let me ask, this is kind of a related question but it's a little bit different because now I'm dropping down into the staffing of the fire brigade but the same kind of issue regarding fire brigade staffing versus other staffing to support operations of the unit.

And again, the reference for the FSAR is Section 9.5.1.6.3. It relates to qualifications of the fire protection personnel.

And that section says, the brigade leader and at least two brigade members have sufficient training in or knowledge of plant systems to understand the effects of fire and fire suppressants on safety shutdown capability. A brigade leader is competent as evidence by possession of an operator's license or equivalent knowledge of plant systems to assess the potential safety consequences of a fire and advise MCR personnel.

And maybe I'm reading too much into this paragraph but it seems to tell me that three of the five fire brigade members are, basically come from the operations organization. Because they're qualifications are knowledge of plant systems,

understand the effects of the fire on safe shutdown capability and things like that.

And I guess I'll just read the questions that I had just so that we kind of get them out on the table. And one of them was, is a license reactor operator or a senior reactor operator always assigned to be the fire brigade leader?

I mean that's just a basic question. They're implications that it might be but of that's the case, I'm not sure about staffing in the main control room.

If you don't have a license reactor operator or an SRO as the fire brigade leader, how does the fire brigade, how do you ensure that the fire brigade leader has the qualifications as stated here that says, is competent as evidence by possession of an operator's license?

So that's clear that it is a licensed operator or equivalent knowledge of plant systems. That means an unlicensed person who has equivalent knowledge to a licensed operator.

MR. EVANS: But with knowledge of plant systems.

CHAIRMAN STETKAR: The plant systems to assess the potential safety consequences of a fire and

advise main control room personnel. So it's not just, yes indeed this pump puts water from that tank to that injection line it's, gee, if I burn up this pump, what implications does that have on overall plant safety response. Which is a higher level than, for example, the knowledge that typical equipment operators would have.

And then the final question I had is, is if indeed three members of the five member fire brigade are basically gleaned from the operations staff, the on shift operation staff, how does that effect the availability of licensed operators in the control room, whether it's an RO or an SRO, and the number of equipment operators that are left not responsibly for fire brigade duties to actually manipulate equipment in the plant?

And as I said, some of these questions, the responses to some of those questions hinge a little bit on the fact of whether or not you're going to have a shared fire brigade between the two units or whether you have two units specific fire brigades. I can see a shared fire brigade that has benefits and determents.

The shared fire brigade you can start pulling people from the other unit --

MR. EVANS: Right.

CHAIRMAN STETKAR: -- to fulfill some of these functions. Unit specific fire brigades tend to

start getting pretty thin on being able to staff the fire brigade responsibilities, at least as the way they're outlined here, and maintain sufficient staffing to fulfill everything else that needs to be done at the same time.

Now you probably don't have, unless you really thought a lot of this stuff in detail you may not have answers right now, but I'm just trying to raise the question in terms of looking forward in terms of overall staffing and sort of the plans to define and man and identify responsibilities for a fire event.

MR. WOODLAN: I think you had it right when you were discussing the fact that we're still working on this.

CHAIRMAN STETKAR: Okay.

MR. WOODLAN: It is a level of detail lower than what's in the FSAR.

CHAIRMAN STETKAR: It is.

MR. WOODLAN: You may remember that with Tim Clouser was here --

CHAIRMAN STETKAR: Yes.

MR. WOODLAN: -- who discussed this last time, former operations manager and very deeply involved with this, and he quoted the, he identified what they do in Unit 1 and 2 --

CHAIRMAN STETKAR: Yes.

MR. WOODLAN: -- and it's actually part of the watch bill.

CHAIRMAN STETKAR: Yes.

MR. WOODLAN: And before every watch staffing they go over it and designate it on there are who are the fire brigade members and they assure that those are sufficiently separated, all the other activities can be performed. As I recall, his answer on 3 and 4 was he suspected it was going to be the same.

The biggest difference is 1 and 2 have a lot of common equipment therefore it really needed to be a single fire brigade.

CHAIRMAN STETKAR: Yes.

MR. WOODLAN: 3 and 4 doesn't need to be a single fire brigade.

CHAIRMAN STETKAR: Right.

MR. WOODLAN: It will be an administrative decision which hasn't been made yet.

CHAIRMAN STETKAR: I know we had the discussion during Chapter 13 at sort of that level.

MR. WOODLAN: Yes.

CHAIRMAN STETKAR: Now in Chapter 9 they're kind of drilling down into more specifics about qualifications of individual responsibilities, this

incident commander responsibility that the fire brigade members themselves, in terms of their interface and knowledge of equipment operations and the types of recommendations that the fire brigade leader would be expected to make.

So that's now starting to drop down into that additional level of detail where providing some assurance that indeed all of the functions will be fulfilled, if it is a shared fire brigade between the two units, if it's a unit specific fire brigade for that particular unit in an event of a fire, it can be pretty important.

You know, we've had real events where lack of kind of cleared guidance in terms of responsibilities has led to confusion, if nothing else. And yes, I understand this notion of assembling people at the beginning of this shift and saying, A, B, C, D, E, you're the fire brigade today and, A, you're the fire brigade leader which is all fine if I can select people from operations and maintenance.

And then I think it was even mentioned, security staff might in some cases be assigned to the fire brigade --

MEMBER SCHULTZ: And training staff might be as well.

CHAIRMAN STETKAR: -- and training staff if the training staff was there.

MR. WOODLAN: And it is a little more complex than that.

CHAIRMAN STETKAR: Yes.

MR. WOODLAN: I mean you probably realize is there are specific training programs for various roles in the fire brigade.

CHAIRMAN STETKAR: Yes.

MR. WOODLAN: We do maintain a qualification list. And one of the jobs of the shift coordinator, it's probably the shift manager, is to confirm that any individual assigned a task does in fact have active valid qualifications.

The training programs and the frequency of retraining are all based on assuring that that happens.

CHAIRMAN STETKAR: Okay.

MR. EVANS: I think I agree. It is very important that all the functions are identified and they're planned out and to make sure that you have the staffing required with the proper qualifications to cover those functions.

And absolutely as we developed the details of the program we will ensure that that happens and --

CHAIRMAN STETKAR: And of course the

concern, you know as I read these things, as you sort of focus on each section of the FSAR and isolation, it's easier to write words that says, yes, we're going to have somebody on the fire brigade that will do A and B and C because it's important to do that.

MR. EVANS: Right. The cumulative, your concerned about the --

CHAIRMAN STETKAR: The cumulative will effect --

(Simultaneous speaking)

CHAIRMAN STETKAR: -- when you start looking at the number of bodies you actually have on shift.

MR. EVANS: And the cumulative effect will --

CHAIRMAN STETKAR: And that's where people get in trouble.

MR. EVANS: Yes.

CHAIRMAN STETKAR: You know, quite honestly that's where we've seen issues in the past where people that had dual responsibilities and had to make a decision about, you know, do I go support the fire brigade or do I go operate a piece of electrical equipment because I'm the guy whose supposed to do both.

MR. EVANS: Yes, understand.

CHAIRMAN STETKAR: Okay. That's all I had on the fire brigade. That's enough.

I did have one, since we are talking about the, everything with the fire protection, right, this is the only slide on fire protection?

MR. EVANS: That's correct.

CHAIRMAN STETKAR: In the, of course I've lost my notes here, in the fire hazards analysis, and I'll give you the alphabet soup again, it's Appendix 9A, Section 9A.3.113 which is just the general fire hazards analysis. In particular the fire area that I'm looking at is FA7-301. It's the transformer yard.

MR. EVANS: Yes.

CHAIRMAN STETKAR: And I understand how it was divided up. There's essentially a separate fire area for each of the transformers out there, which makes sense, and I understand the separation, I understand their protection.

There was a statement though there that says, since none of the four safety trains of equipment provided to assure plant shutdown would be effected, no adverse impact of safe shutdown would result from a fire in the transformer yard. Reserve Auxiliary Transformers 3 and 4 are the offsite power supplies to all of the safety buses.

MR. EVANS: One of the offsite power supplies, correct.

CHAIRMAN STETKAR: One of the offsite power supplies.

MR. EVANS: Yes.

CHAIRMAN STETKAR: And I was curious, I don't know the nuances of the word when it says, no adverse impact on safe shutdown would result. Certainly if I burn up one of those transformers I'm going to have demands on things.

But if by you mean I'm not going to disable safe shutdown ability, I totally agree with that. You do have other power supplies around.

MR. EVANS: Right, obviously the loss of reserve auxiliary transformers, for whatever reason, whether it's due to a fire or due to some other reason, would cause a transfer for the safety buses to the alternate power source. Yes.

CHAIRMAN STETKAR: Is that the typical, the concern that I have is people doing fire analysis just making words that says, well this is not a safety related piece of equipment so I don't care about it. When in the real world the plant kind of really cares if you burn up one of those transformers.

MR. EVANS: Absolutely.

CHAIRMAN STETKAR: I mean plant equipment plus not to mention the plant people, but --

MR. EVANS: Yes.

CHAIRMAN STETKAR: I'm just kind of hung up on this sense of the glib statement that it wouldn't have any adverse effect. And if by that the implication is it would not prevent safety shutdown, I fully agree with that.

MR. EVANS: Yes, I think that is the implication.

CHAIRMAN STETKAR: Okay. Thanks. Again, sometimes I hang up on words because I've seen people who are experts in fire protection programs and experts in fire analysis who don't understand at all how a plant works.

MR. EVANS: Right.

CHAIRMAN STETKAR: And they sometimes make very simplistic statements about the fact because they're given a list of "safety related equipment" and as long as that list doesn't exactly match some of the pieces of equipment that they're looking at, they check off a box that says, it's not important at all.

Okay, thanks. I've got. Again, I just wanted to kind of raise that because it was the only place in the whole fire hazards analysis where I actually

stumbled across something that was a bit vague.

MR. EVANS: Okay.

CHAIRMAN STETKAR: Thanks. You can flip your slide now.

MR. EVANS: Okay.

CHAIRMAN STETKAR: You've been waiting long enough, interpret five seconds of silence as flip the slide.

MR. EVANS: Moving right along. Let's move to communications, Section 9.5.2.

There are various intra-plant and offsite communication systems provided for both normal and emergency conditions. External communication links include things such as standard telephones, microwave telephones, fiberoptic links for fiber optic data, radios, direct telephone links, i.e., ring down type lines to state local NRC, satellite telephones and of course personal cell phone.

Plant communications are backed up by eight hour battery capacity. And I think that covers the communication system aspects.

Fuel or storage tanks are discussed in Section 9.5.4. These are site specific aspects.

The underground vaults containing the fuel oil tanks for the gas turbine generators are provided.

They provide a seven day supply of fuel oil.

Each vault has a vapor and liquid detection devices. Each fault has manually operated ventilation and temperatures, hang on a second, and temperatures in the vault will not go below the fuel oil cloud point.

Vault temperatures can go down to 32 degrees but the fuel oil in the tanks will not go below the fuel cloud point. And that finishes the open part of our presentation for Chapter 9.

CHAIRMAN STETKAR: Let me ask you one question that is kind of an it at the interface. And I don't know whether it's better to talk about it now or later.

As I understand it, the essential service water system is designed, there are cross connects between essential service water and the fire protection system for the reactor building and the essential service water pump house. Such that during a safety shutdown earthquake at least, I mean that's kind of an nominal event but it could be other conditions, you can manually align essential service water to supply at least a couple of host stations in each location.

There's some 75 gpm design and 18,000 gallons of capacity.

MR. EVANS: Right.

CHAIRMAN STETKAR: I'm not so concerned about the firefighting of water lines or things like that. What I was interested in, and this is another one of these off the wall questions, because those supplies are aligned manually through local manual valves, are those local valves located in the areas where you need to fight the fire?

Because here's the thought. I have a fire, I can't use my normal fire protection system. Before I can start fighting the fire I need to send somebody into the place where the fire is burning to open the manual valve so that I then get water to my fire hose so I then can start putting out the fire.

I don't want to be the first guy going in there to operate that valve. I'd rather have the valve outside of where the fire is burning or hoses working in an area where if I need to go manipulate equipment there.

So I was curious whether you'd thought about it that much? The only reason I ask is on all the drawings, some of the drawings seem to say that they're inside the fire areas, most of the drawings shown them at the border between two locations.

And I'm sure you don't have that detail yet, maybe you do. But it's something to think about.

MR. EVANS: Yes, we can check into that.

The valve that is the, coming off of the service water line where the tap is that feeds into the fire protection system is in the, to my knowledge, is in the ultimate heat sink area. So --

CHAIRMAN STETKAR: Well there's two of them. There's one --

MR. EVANS: But there could be other valves, yes.

CHAIRMAN STETKAR: -- there's one that goes, I mean there's actually two branch lines. There's one that feeds, and I don't know what fire hose risers there are, but there's one that feeds part of the reactor building and there's another one that feeds the essential service water pump rooms.

MR. EVANS: That's what it was.

CHAIRMAN STETKAR: And certainly the one that feeds the essential service water pump rooms is someplace in the ultimate heat sink building.

MR. EVANS: That's the one I was referring to, yes.

CHAIRMAN STETKAR: Okay. But the other one for the reactor building only shows up, it typically shows up at a border between the reactor building and other locations.

MR. EVANS: It's a P&ID --

CHAIRMAN STETKAR: It's even worse, it's got an artist conception.

MR. EVANS: Yes.

CHAIRMAN STETKAR: And again, it's a subtle nuance but if you're developing a fire protection plant, this is more supposed to than ESW obviously. I don't care whether the alternate supply is coming from ESW or whether it's coming from a pumper truck.

MR. WOODLAN: I can't answer exactly where those --

CHAIRMAN STETKAR: Oh.

MR. WOODLAN: -- are but I can tell you we had that discussion.

CHAIRMAN STETKAR: Oh, really?

MR. WOODLAN: I asked the same question and they showed, and as I recall and I may have this wrong, but I believe one of the valves, for example, was located in a stairwell away from the fire area that we're talking --

CHAIRMAN STETKAR: That's exactly the kind of --

MR. WOODLAN: So we did look at it although I can't tell you exactly where they are.

CHAIRMAN STETKAR: That's all I wanted to hear is you already thought about it.

MR. WOODLAN: Okay.

CHAIRMAN STETKAR: Thank you. Thanks.

MR. EVANS: Well that completes the open portion. If no further questions we can move to the --

CHAIRMAN STETKAR: Do any of the members have any questions for Luminant as far as the material that we've covered in the open session?

MEMBER RAY: No.

CHAIRMAN STETKAR: If not we will now close the meeting so that we can discuss the closed information regarding the ultimate heat sink and essential service water system.

(Whereupon, the foregoing matter went off the record at 9:20 a.m. and went back on the record at 9:49 a.m.)

CHAIRMAN STETKAR: I believe that Luminant is now done with your presentation completely, so I'd like to, thank you very much, you covered a lot of material. And again, just to make sure that we have everything covered, do any of the members have any other questions for Luminant?

MEMBER SCHULTZ: No, sir.

CHAIRMAN STETKAR: If not, thank you, I appreciate it and what we'll do is we will take a break so that we don't have to interrupt the staff.

MR. SHUKLA: The staff is the only one that is late.

CHAIRMAN STETKAR: We will take a break so that we don't have to interrupt the staff and we will reconvene at, I'll be generous, ten minutes after 10:00.

(Whereupon, the foregoing matter went off the record at 9:50 a.m. and went back on the record at 10:11 a.m.)

CHAIRMAN STETKAR: We are back in session. Just for the record this is an open session and we will hear from the staff on their review of COL Chapter 9.

MR. KALLAN: Okay.

CHAIRMAN STETKAR: Paul?

MR. KALLAN: Thank you very much. My name is Paul Kallan. I'm the Senior Project Manager and also the Chapter P.M. for Chapter 9. Steve Monarque is the lead for this project and Larry Wheeler is the technical staff member.

We had one open item on the SER, it's basically the DCD made a change in the calculations and Commanche Peak has to reflect this because it's more of a documentation and Larry's going to explain more.

CHAIRMAN STETKAR: Okay.

MR. WHEELER: Good morning. Like Paul had mentioned earlier, or just now, and the Applicant had

mentioned earlier, this is more of a documentation issue.

This is to catch up the Applicant's FSAR to the DCD. DCD Rev 4 indicated there are some small changes for the heat load, safe shutdown with the LOOP peak heat load remains the same.

The LOCA peak load did have a small 2 percent change. The LOCA 30-day heat load for development of heat sink slightly affected the 30-day cooling capacity.

These changes are going to be shown in, or picked up by the Applicant in their FSAR Section 9.2.5.2.3. With that said, the next two lines show the small changes going from 8.4 million to 8.3 for the safe shutdown in the LOOP.

And then for the LOCA it goes from 8.2 to 8.4 million for the LOCA. So the concerns the staff has are none. The 8.4 remains the same even though --

MALE PARTICIPANT: Right.

MR. WHEELER: -- how you got there came from a different heat load.

MEMBER SCHULTZ: What accounts for the decrease in heat load related to the shutdown with LOOP evaluation? I mean it sounded like in the description that nothing had changed except the answer.

MR. WHEELER: I don't have that information, that would come from the MHI, the D.C. What

I can say, that I did an audit of the calculations, Comanche Peak, and when I got into the details of this calculation it was, the staff, with MHI and Comanche Peak staff was there, had said there were some changes that were coming down the road.

That the calculations I had saw at that time were in fact going to change slightly and now this is the information that they are going to bring forward for us to look at.

MEMBER SCHULTZ: Okay. I understand. Thank you.

CHAIRMAN STETKAR: Larry --

MR. WHEELER: If you flip to the next slide we're essentially done with my presentation, but I'm open to questions.

CHAIRMAN STETKAR: No you're not.

(Laughter)

CHAIRMAN STETKAR: I did have, were you in earlier when I asked the Applicant about the calculation for the minimum heat removal, the meteorological conditions for minimum heat removal versus maximum evaporate loss?

Am I misunderstanding something or --

MR. WHEELER: Well the --

CHAIRMAN STETKAR: -- do you have any help?

MR. WHEELER: You're going in the right path. I did look at the calculations at the audit, I can't remember how long ago that was, but there was a set of calculations that looked at return temperature, because that's what we're concerned about is --

CHAIRMAN STETKAR: Yes.

MR. WHEELER: -- and is hot water going to the cooling tower and cold water going back to the plant, that's a different set of calculations than the evaporative cooling calculations.

CHAIRMAN STETKAR: Right. Because for the calculation with the return temperature you want to --

MR. WHEELER: Maximum wet bulb.

CHAIRMAN STETKAR: Yes, maximum wet bulb.

MR. WHEELER: Yes.

CHAIRMAN STETKAR: The other one you want maximum, isn't absolute minimum what I, you have the --

MR. WHEELER: But you're also looking --

CHAIRMAN STETKAR: -- maximum difference between the wet and dry bulbs.

MR. WHEELER: -- at a duration of 30 days.

CHAIRMAN STETKAR: Yes.

MR. WHEELER: Now a duration of 30 days you're now looking at an accident --

CHAIRMAN STETKAR: Yes.

MR. WHEELER: -- because you're looking at the maximum heat load.

CHAIRMAN STETKAR: Yes.

MR. WHEELER: So what I had done and I know I might be getting into some proprietary information, if I get too deep into the calculation --

CHAIRMAN STETKAR: And if you do we can close the meeting.

MALE PARTICIPANT: Right.

CHAIRMAN STETKAR: No problem at all. So it's up to --

MR. WHEELER: Let me go down this path, I think I'm going to be okay, is I looked at the calculation, I made a lot of good notes, I brought those notes back and I essentially made my own Excel spreadsheet.

And that Excel spreadsheet pretty much resulted in the same outcome that MHI and Comanche Peak came up with, and the concern was, is how much water do I need for 30 days for the design basis accident.

So I made all these columns as you can see, there's lots of columns, lots of calculations, I let Excel spreadsheet do all that math, and I came up with exactly the same results that I saw at the calculation at Comanche Peak.

What I'm trying to say is, the input into that calculation fixed the return temperature at 95 degrees, and in order to fix a return temperature of 95 degrees the cooling tower efficiency has to be taken into account for wet bulb.

So you have to assume, because now you're talking a 30-day duration, accident conditions, you have to fix something, 95 degrees, and then you work through your calculation.

CHAIRMAN STETKAR: But if I took a temperature profile, if I took, I mean, what I know is the accident heat load --

MR. WHEELER: That's correct.

CHAIRMAN STETKAR: -- that's something that's fixed --

MR. WHEELER: And it's always changing. It takes after a couple hours and then it tails right off.

CHAIRMAN STETKAR: It's, you know, it's a changing load, but I nominally know that.

MR. WHEELER: Right.

CHAIRMAN STETKAR: And if I fix my cooling tower design --

MR. WHEELER: Yes.

CHAIRMAN STETKAR: -- and then change my meteorological, my wet bulb temperature, meteorological

parameters --

MR. WHEELER: Okay.

CHAIRMAN STETKAR: -- and allow the return temperature, the basin return temperature to vary, such that the meteorological parameters that I set get me the maximum evaporative heat loss --

MR. WHEELER: Okay.

CHAIRMAN STETKAR: -- would I come up with a different basin inventory requirement? Follow me?

MR. WHEELER: Time to go to one of the backup slides and let me tell you what slide that is, Slide 7.

CHAIRMAN STETKAR: See I told you it's longer than one slide.

MR. WHEELER: That's four.

CHAIRMAN STETKAR: You didn't believe me.

(Laughter)

MR. WHEELER: Backup slides, there we go. So this is the information related to how much volume I need on site for 30 days. So you start with the Reg Guide, prior 30-day cooling.

They looked at the meteorological conditions for the worst case, 30 years of data --

CHAIRMAN STETKAR: Now define for me carefully what "worst case" means. Does that mean the

minimum difference between the wet bulb and dry bulb over a 30-day period?

MR. WHEELER: No, this is just looking at non-coincident wet bulb.

CHAIRMAN STETKAR: Okay. But --

MR. WHEELER: If they took that data --

CHAIRMAN STETKAR: But the highest non-coincident wet bulb.

MR. WHEELER: Right.

CHAIRMAN STETKAR: Okay.

MR. WHEELER: And they came up with an average wet bulb for 30 days.

CHAIRMAN STETKAR: Yes.

MR. WHEELER: Okay. Now what they did on top of that, so you're starting with 78 --

CHAIRMAN STETKAR: Right.

MR. WHEELER: -- so obviously you want some margin --

CHAIRMAN STETKAR: Right.

MR. WHEELER: -- so you build in some margin, the margin looks at a recirculation --

CHAIRMAN STETKAR: Right.

MR. WHEELER: -- two degrees, it's reasonable, and then they turn around and add an additional three degrees --

CHAIRMAN STETKAR: Right.

MR. WHEELER: -- so now you're at 83.

CHAIRMAN STETKAR: Right.

MR. WHEELER: So they calculations they all looked at had a starting point of worst case of web bulb of 83 degrees.

CHAIRMAN STETKAR: Right.

MR. WHEELER: So you work through all those calculations to verify that you had a return temperature of 95.

CHAIRMAN STETKAR: Right, given --

MR. WHEELER: Ninety-five is now the input into the evaporative cooling, because now you're looking at worst conditions, DBA or LOCA, over 30 days the worst meteorological conditions per the Reg Guide.

So you're starting at 83, going to get a return temperature of 95 and then now you fix that 95 degrees for 30 days, that's being a very conservative analysis, 30 days of 95 degrees and looking at the heat load peaks and then the degraded heat load over 30 days, so you come up with this huge volume as a worst case.

CHAIRMAN STETKAR: I guess what I'm asking, maybe I'm still not understanding the calculation and if I get too dense just tell me and we'll move on.

Suppose that they had selected, they looked

at the 30-day period and selected an average wet bulb temperature of, pick a number, 65 degrees.

MR. WHEELER: Okay.

CHAIRMAN STETKAR: Given the LOCA heat load --

MR. WHEELER: Okay.

CHAIRMAN STETKAR: Don't fix the 95 degrees, just take the LOCA heat load, would that have affected the total amount of evaporative heat loss such that the basin capacity requirements would be more than under this calculation? Follow me?

MR. WHEELER: The deficiency of a cooling tower is --

CHAIRMAN STETKAR: Related to the temperature, yes.

MR. WHEELER: -- is related to the wet bulb which you're going to get --

CHAIRMAN STETKAR: One.

MR. WHEELER: -- a delta between wet bulb and return temperature of somewhere around, we'll just use the number eight to ten.

CHAIRMAN STETKAR: Right.

MR. WHEELER: So if you're wet bulb is at 65, and we'll just round it up to 75, now ESW is at 75. I did not look at doing any calculations at 75 degrees

starting point.

CHAIRMAN STETKAR: What I don't understand, because I don't do these kind of equations is whether the --

MR. WHEELER: But I can certainly go through what I did here and input different values --

CHAIRMAN STETKAR: I mean I see --

MR. WHEELER: -- and see what the results, but --

CHAIRMAN STETKAR: I have seen people do those two different, they'll do this one which essentially defines the volume and the cooling tower efficiency to maintain temperatures less than or equal to 95 degrees in the basin.

MR. WHEELER: Sure.

CHAIRMAN STETKAR: And I've seen people do a separate analysis that, like I said, fixes the heat input and looks at a temperature profile where you essentially maximize the difference between the wet bulb and dry bulb temperature to see if that makes any difference in the required basin inventory, in other --

MR. WHEELER: Right.

CHAIRMAN STETKAR: -- maximize the evaporative heat loss, but let the return temperature, you know, vary to whatever it would be.

MR. WHEELER: Now what I can say is when we walked in and did the calculation reviews for Comanche Peak, you know, they give us the calculations of concern.

They might have done other calculations to rule out what you were just saying and then just present to the NRC, this is the most bounding condition.

CHAIRMAN STETKAR: Yes. I just wanted to make sure -- yes.

MR. WHEELER: So we would have to pull that thread and go back to Comanche Peak because I, are there other supporting calculations that now indicate what you gave us is now bounding.

CHAIRMAN STETKAR: Yes. And that's what I, I asked that, you know, in the earlier session and they said --

MR. WHEELER: And I think then I could, at the same time I can go back and crank through, you know, a whole bunch of numbers at different temperatures.

CHAIRMAN STETKAR: But I mean cranking through numbers, and it still has to be realistic for their site --

MR. WHEELER: Sure.

CHAIRMAN STETKAR: -- I mean they selected this set of temperatures with their penalties based on their review of 30 years of meteorological data for that

site.

So it isn't, you know, that 78 degree temperature that they started isn't a hypothetical temperature --

MR. WHEELER: Correct.

CHAIRMAN STETKAR: -- like my, you know, I picked 65, I really don't know, you know, what the appropriate temperature distribution would be for that type of calculation.

MR. WHEELER: But once again for this site we do have four, 33 2 percent basin sitting there, and the fact that the water level gets low you got, you got too many of gallons sitting and the next --

CHAIRMAN STETKAR: No, I, you know, I understand, too, also.

(Simultaneous speaking)

CHAIRMAN STETKAR: I also do.

MR. WHEELER: There's margin built into this site that should not get them into trouble.

CHAIRMAN STETKAR: Yes.

MEMBER SCHULTZ: But beyond that given what they're working to achieve in terms of the 30-day response, the depiction of the site in high temperature --

MR. WHEELER: Correct.

MEMBER SCHULTZ: -- 78 plus five degrees to the delta of 95, that would appear to be the bounding calculation.

MR. WHEELER: Yes, as what I see and how it appears to be done.

CHAIRMAN STETKAR: It may very well be through this site. I mean it's not Palo Verde, for example, so there is, you know, in terms of evaporative losses.

MR. WHEELER: So what you're saying, you want the staff to further look at this, I mean like --

CHAIRMAN STETKAR: Well I asked, I mean --

MR. WHEELER: -- Comanche and MHI to --

MEMBER SCHULTZ: We put it out on the table. I mean I'd like some assurance that somebody at least thought about --

MR. WHEELER: Sure.

MEMBER SCHULTZ: -- that the calculation that's presented in the FSAR as, the way it's presented is these are the worst possible conditions for everything and indeed the design satisfies the worst possible conditions.

MR. WHEELER: Yes.

MEMBER SCHULTZ: And I guess we're asking for assurance that indeed, not only the medium cooling

requirements if I can cast them that way --

MR. WHEELER: Sure.

MEMBER SCHULTZ: -- but the maximum evaporative heat loss --

MR. WHEELER: Sure.

MEMBER SCHULTZ: -- essentially maximum, the best possible cooling conditions based on actual, you know, site data, were also considered. Whether that would make a substantial, any difference in the decisions about either inventory or timing of the need to transfer water from basin to basin or things like that.

MR. WHEELER: Okay. I understand.

MEMBER SCHULTZ: Okay. Thanks.

CHAIRMAN STETKAR: You did that very well without getting into proprietary stuff.

MR. WHEELER: Thank you.

CHAIRMAN STETKAR: The other question that I had, and this more of a not-so-technical question, is that the FSAR, I know they're doing, you know, they refer to Reg Guide 1.189 and NFPA-804 as their basic references for guidance and criteria.

Do you know whether they've, and I'm not sure whether I know this or not, are they complying, complying is the wrong word, are they using Reg Guide 1.189, Rev 2 or are you reviewing them to Reg Guide 1.189,

Rev 2 because, you know, there's substantial changes to Rev 2.

MR. WHEELER: Right. Yes, I understand your question and let me see if there's someone in the audience that's a fire protection engineer.

CHAIRMAN STETKAR: And the other part of the question is, NFPA-805 seems to apply for Part 52 licensees, and I'm not familiar with 804.

MR. WHEELER: Okay.

CHAIRMAN STETKAR: There's no reference to NFPA-805 whatsoever in the FSAR. NFPA-805, you know, obviously, can be used for deterministic, it doesn't have to be used, the rule in NFPA-805 gives you the option to have a risk-informed fire protection program, but it also contains guidance and requirements for a deterministic.

So I was curious whether or not the fire protection program design, let me get the full question out, is being reviewed in the context of Reg Guide 1.189, Rev 2, and NFPA-805?

MR. WHEELER: I think I'm going to get some help on this one.

CHAIRMAN STETKAR: And now we see --

MR. DIAS: Let me say something. This is Antonio Dias. The person, the expert of the technical staff responsible for our protection is not here today.

We're going to get back to you on this later, hopefully in two weeks.

MR. WHEELER: Okay.

CHAIRMAN STETKAR: Great. Thank you.

MR. DIAS: Thank you.

CHAIRMAN STETKAR: I don't have anything more for the staff. Do any of the other members have anything more for the staff?

(Multiple no's)

CHAIRMAN STETKAR: If not, what I'd like to do before we transition to MHI's responses to some open items from yesterday and additional information from the staff, is I'll ask first if we have any public comments from anyone in the room regarding the Chapter 9 review.

MR. MONARQUE: Hey, John?

CHAIRMAN STETKAR: Yes.

MR. MONARQUE: Let me, you had asked a question about which revision of Reg Guide 1.189 --

CHAIRMAN STETKAR: Yes.

MR. MONARQUE: -- maybe Luminant can answer that question to you, if you're interested? They would know --

CHAIRMAN STETKAR: Luminant can.

MR. EVANS: This is Todd Evans with Luminant. Yes, both the DCD and the COLA are to Reg Guide

1.189, Rev 1.

CHAIRMAN STETKAR: Rev 1. Okay, now, then my question to the staff will be, going forward, since 1.189, Rev 2 has been out since 2009, I think, and NFPA-805, I've forgotten the dates on the rule that endorses 805 is, going forward, what do we do, or what do you? I don't do anything.

MR. MONARQUE: And we understand your question. We'll get back to you on that.

MR. WHEELER: Okay.

CHAIRMAN STETKAR: But it is, just for the record, it is 1.189, Rev 1?

MR. WHEELER: Yes.

CHAIRMAN STETKAR: Okay. I thought that I remembered that, but I didn't, yes, now I found the citation. It is 1.189, 1. Okay, thank you.

MR. MONARQUE: And I think we listed the SE.

MR. WHEELER: Yes, we did.

CHAIRMAN STETKAR: Yes, and in fact I just found it. I have so much notes that sometimes I get lost. It is cited in Table 9.5.1-1R. They make a comparison, point-by-point comparison, it's a long Table, point-by-point comparison, it does specifically cite Rev 1 of Reg Guide 1.189.

So going forward since Rev 2 has been out

now for, it was issued in October 2009, so we're now four years into that.

MR. WHEELER: Okay.

CHAIRMAN STETKAR: Going forward, and there are differences as you well know in terms of things like treatment of multiple spurious operations and some other subtleties.

MR. WOODLAN: John, this is Don Woodlan. Did you want confirmation on 805?

CHAIRMAN STETKAR: Well 805 is not mentioned anywhere.

MR. WOODLAN: Right. We did not choose that option.

CHAIRMAN STETKAR: Yes.

MR. WOODLAN: We are not using 805.

CHAIRMAN STETKAR: Well, okay, but it's my understanding, I've kind of been through this, 805 is not, everybody thinks of NFPA-805 as risk informed fire protection because that's the context that everybody speaks of it.

NFPA-805 outlines a fire protection program and it gives you an option of doing a risk informed process, but that's an option.

MR. WOODLAN: Yes.

CHAIRMAN STETKAR: NFPA-805, you can

develop a deterministic, if I want to call it that, fire protection program following the guidance in NFPA-805 without ever saying those nasty words PRA.

MR. SHUKLA: It allows to cherry pick those.

CHAIRMAN STETKAR: It allows you to cherry pick, you can mix and match deterministic and probabilistic. I'm not familiar with 804, so I don't know how 804 and 805 work together or whether there are any discrepancies between those two in terms of the requirements that they contain.

You know, the 10 CRF 50.48 does not mention NFPA-804. It's not mentioned. It only mentions NFPA-805 as being the endorsed standard. So I'm curious how we're, you know, transitioning in time here in terms of the requirements and whether reliance on older requirements could be missing some things that we ought not to miss.

MR. WHEELER: Okay.

CHAIRMAN STETKAR: As I said, that's all I had. I'm not sure whether I was interrupted or not, but again, just for the record, I'll ask are there any public comments regarding the review of Chapter 9?

Hearing nothing in the room, we don't have the public on the bridge line, so what I'll do is, again,

thanks a lot for the staff. One slide only took 25 minutes so that was pretty good and thanks again. I appreciate it.

With that, let's, I'll put MHI off. The staff wanted to present some information that would help, I believe, elaborate on some of the discussion that we had yesterday, so let's ask them to come up first and do that.

MR. GALVIN: And, John, we decided we're not going to do that at this point.

CHAIRMAN STETKAR: Okay. Thank you. That was quick. In which case, Ryan, do you have some information that you'd like to present to -- or they have the answers to some of the issues we raised yesterday?

MR. SPRENGEL: Are we open or closed?

CHAIRMAN STETKAR: We are open. We're still, unless MHI, do you need to close it for anything for this?

MR. SPRENGEL: No, open is fine. We'll need to connect.

CHAIRMAN STETKAR: Yes, okay, so --

MR. SPRENGEL: We'll have to talk to them.

CHAIRMAN STETKAR: -- we'll get that set up somehow.

(Pause)

MEMBER BROWN: Do you need Theron to do his connection?

CHAIRMAN STETKAR: I certainly do because I have no idea what this is about.

MEMBER BROWN: No, I was just -- he was looking for cables. I don't --

MEMBER BROWN: Looks like he wanted to find cables, so that's all.

MALE PARTICIPANT: Just grab one.

CHAIRMAN STETKAR: I'm lucky I can turn a light on without electrocuting myself much less hook up cables.

(Pause)

CHAIRMAN STETKAR: Theron, help.

MEMBER BROWN: I think I lost my electrical safety lesson for the one time that I learned when I was eight years old, in other words, don't stick a screwdriver with a wire in one part of the socket, stick the screwdriver in the other one and watch part of the screwdriver disappear.

I thought that was really cute at eight years old, fortunately it was an insulated handle. I carried that with me, it got lost in when electricity is dangerous.

CHAIRMAN STETKAR: It's, you know, Darwin

tends to take care of some of these things. It's like --

MEMBER BROWN: I built a radio, a 5-tube radio, and it said strip, you know, take all the insulation off --

CHAIRMAN STETKAR: We're on the record by the way.

MEMBER BROWN: Oh, okay. Thought we were waiting here.

CHAIRMAN STETKAR: We are, but we wait on the record.

MEMBER BROWN: Oh, that's okay. I don't mind being embarrassed. Tell him not to record my story.

CHAIRMAN STETKAR: Too late.

(Pause)

CHAIRMAN STETKAR: I'm starting to believe that we need a few little buttons. Sam and I have talked occasionally, the little light to tell us when the bridge line is open.

I think we need a little switch to put the Muzak on in the background during these pauses.

MALE PARTICIPANT: Is Theron coming or what?

MR. SHUKLA: He's coming.

MALE PARTICIPANT: He's coming, okay.

MALE PARTICIPANT: There we go.

MR. SPRENGEL: Okay. To not spend anymore time waiting on that, we do have a list of --

CHAIRMAN STETKAR: We should be up. Are you?

MR. SHUKLA: Yes, we are up.

MR. SPRENGEL: Yes. He'll just need to open up those specific areas, but we have about 12 items that we've --

CHAIRMAN STETKAR: Oh.

MR. SPRENGEL: -- identified from yesterday to follow up on to differing levels of detail. I think they are all pretty clear. And now today we'll try to go through a couple of the items and if we're able to resolve them that's great.

If not, I think it'll give us an opportunity to kind of refine the question --

CHAIRMAN STETKAR: Yes, that's great.

MR. SPRENGEL: -- and get to the real heart of the question.

CHAIRMAN STETKAR: That's great.

MR. SPRENGEL: And I think some of them yesterday were leading to real issues that we really didn't get to. So we'll go through a few items now and oddly, but fortunately, we have ample time to go through

those.

So I'll turn it over to Nagai-san to introduce the first topic.

MR. NAGAI: Thank you, Ryan. Good morning everyone. This is Masatoshi Nagai with MNES. There were about, you have open questions from yesterday's discussion and we'd like to cover as many items as possible today.

I'd like to start with Section 3.2 and move forward in the section order. So the first question was, the feed water system is considered to be a risk-significant system in Table 17.4-1, but in Table 3.2-2 it is not classified accordingly and you were asking why that was.

And Nishio-san is going to provide an answer for that.

MR. NISHIO: I am Hiroki Nishio of MHI. Yes, I confirmed that Table 17.4-1 and then either one -- that's, it is considered -- and according to our classification 4-C, it becomes a different Class IV or V, it is a discontinued class.

So now coming to DCD we described equipment of Class VIII, so it is not consistent so we've directed to change that, change that feed water classification, you know, it should be in a Class IV to be consistent with

the Table 17.4-1.

CHAIRMAN STETKAR: So just to make sure I understand, you're saying that you proposed them to change the classification from VIII to V for the feed water system.

MR. NISHIO: From VIII to IV.

CHAIRMAN STETKAR: IV, okay. I forgot. I've lost my list of classifications, but anyway, raise it's classification from VIII. Thanks for that clarification.

What I'd caution you to think about is that if, and I believe it's true, if the feed water system is included in Table 17.4-1, for the purpose of delivering water to the steam generators --

MR. NISHIO: Right.

CHAIRMAN STETKAR: -- then not only the feed water system equipment itself, but also the condensate system, the turbine building cooling water system, and the non, whatever it's called, the non-essential service water system or the normal service water system, would also be required plus the power supplies because you need electricity, cooling water, and the source of suction water for the feed water pump.

So that, the reason I raised it is that there's a potentially very large increase in the scope

of equipment that you may be raising the classification on if you include the feed water system --

MR. NISHIO: Yes.

CHAIRMAN STETKAR: -- because it's just not the, it's not only the feed water pumps and feed water control valves and such, it has to then logically also include condensate and all of those support systems, which is quite a bit of equipment.

So I appreciate this as a, let me call it a preliminary feedback, but the implications could be fairly significant.

MR. NISHIO: Yes. Let me direct to the stuff to the PRA people.

CHAIRMAN STETKAR: Yes.

MR. NISHIO: And what function is required for this significant component.

CHAIRMAN STETKAR: Yes, be careful. I mean if it were only feed water isolation, for example, it would be different, because then it would only be the feed water isolation valves and perhaps the feed water control valves.

But from the brief description in Table 17.4-1, it was my understanding that feed water was included in that Table for the function to supply water to the steam generators, which means a lot more equipment

would be included in those higher classifications.

So I would just caution you to keep that in mind.

MR. NISHIO: Okay.

CHAIRMAN STETKAR: And make it very clearly understand when you talk to the PRA people or the Expert Elicitation Team, which also had some input into the population of equipment in that Table 17.4-1, that you clearly understand the functions there.

MR. NISHIO: Okay.

MR. NAGAI: Okay. So our next one, while we were discussing seismic classification in Section 3.2 there was a question about whether MHI has considered pending updates to ground motion data and Ogasawara-san has a response to that question.

MR. OGASAWARA: Yes. This is Hikaru Ogasawara speaking from MHI, Civil Engineer. The updates to the ground motion data in CEUS Seismic Source is COLA issue, this is not DCD issue.

And MHI does not consider this update to the ground motion data in CEUS Seismic Source in DCD. And, also, this issue is covered by DCD Section 3.7.1.

MR. NAGAI: If there isn't any --

CHAIRMAN STETKAR: Does --

MR. NAGAI: Yes, understand?

CHAIRMAN STETKAR: Understand, yes. So essentially we'll cover that once we get to the COL cite specific seismic hazard evaluation in three point, whatever it is, seven I guess. Thanks.

MR. NAGAI: Okay, so next one, Section 3.3, there was a question why there is such a large difference in probability between the severe wind speed and hurricane speed, but there's just only a 5-mile per hour difference in the wind speed.

And I would like to ask Ogasawara-san to, again, to present the response.

MR. OGASAWARA: Yes. This is Hikaru Ogasawara speaking. First of all, the reason of selecting ten to the minus two is based on Standard Review Plan 3.3.1 and to select ten to the minus seven is based on the Regulatory Guide 1.221.

And the reason considering for hurricane speed, that probability is ten to the minus seven per year is defined to cover the United States site as defined in this presentation, a 160-mile per hour line is located around here. That covers our potential customer site.

CHAIRMAN STETKAR: Yes.

MR. OGASAWARA: On the other hand, the region of severe wind speed, that probability is ten to the minus two per year is taken conservatively wider than

the region of hurricane.

In this presentation 155-mile per hour is around this area.

CHAIRMAN STETKAR: Yes.

MR. OGASAWARA: So the reasons of there is a large difference in the probability between ten to the minus two and ten to the minus seven even though the wind speed difference is just five miles per hour.

So the difference is, the covering area is different.

CHAIRMAN STETKAR: If I look at the wind speed contours on the right from Reg Guide 1.76, I see contours that progress inland from the coast. If I look at the contours from Reg Guide 1.221, I see contours that progress inland from the coast.

I honestly don't understand the argument. I do understand the argument if you would admit that Reg Guide 1.76 is based on things that are simply made up and are not based on real data, which is more of a staff problem than your problem.

But I will not accept the notion that 155-mile per hour wind has a recurrence frequency of once in a hundred years, and 160-mile per hour wind has a recurrence frequency of once in ten million years. That just does not make any sense.

Mother Nature doesn't work that way. Something on one of these two depictions is wrong. Mother Nature does not work that way. So we'll just leave that one open.

You've referred us to the Standard Review Plan and Reg Guide Sections that you've used, so indeed your complying with the available guidance. This is more an issue for the staff. The staff's guidance is not consistent.

I'll put that on there. It's now on the public record, it is not consistent. We've raised this issue for the last two or three years and it's just an example of where the staff is using inconsistent guidance based on inconsistent concepts of meteorological parameters.

And the problem is, in this particular issue, and again I'll put this on the record, you are making design decisions based on these two values. There are statements in the Design Certification Document that says your turbine building siding is designed to remain intact for a severe wind of 155 miles per hour. That is a specific design decision.

As the implications on costs, structures, and everything, there are also statements that say for protection against hurricane differential pressures,

the siding is designed to blow off at 160 miles per hour because the design must be organized such that you don't have two large differential pressures.

Those are very distinct constraints on design based on these interpretations of a 5-mile per hour difference with a factor of 100,000 difference in the recurrence interval.

And that's the reason, I know that your buildings are designed for 200 and whatever it is miles per hour tornado loads, you know, which is higher than, different type of loading dynamics, but higher than the hurricane.

I know you design for 166-mile per hour hurricane. The question is, having several different design criteria based on numbers that just don't make any sense.

So it's, I'll put it back to the staff, is I'll ask you to justify why you believe 155-mile per hour wind speed for what we'll call a severe wind from Reference X has a frequency of one event in a hundred years where a wind speed of 166-miles per hour from Reference Y has a frequency of ten to the minus seven.

I'd the like the staff now to justify that, so you can get back to us.

MEMBER SCHULTZ: And we appreciate what you

presented today.

CHAIRMAN STETKAR: Yes, we do.

MEMBER SCHULTZ: Because it demonstrates pictorially what is the issue here and what needs to be resolved.

MEMBER RAY: Did you want to say something more?

MR. SPRENGEL: No. I think we're done at this time.

MR. GALVIN: I think that the Reg Guides may have been on the wrong figures, but I guess it still doesn't change your point. The hurricane is Reg Guide 1.221 --

CHAIRMAN STETKAR: Yes, that's correct.

MR. GALVIN: -- the straight line winds are Reg Guide 1.76 --

CHAIRMAN STETKAR: Right.

MR. GALVIN: -- looks like they had those reversed.

CHAIRMAN STETKAR: On the slide, yes. That's fine.

MR. GALVIN: But your point is well taken.

CHAIRMAN STETKAR: Okay. Thanks. It's only something, and I honestly don't know what difference it makes in the structural design effort or in the review

effort for that matter, except for the fact I did come across that one place that says, you know, for a 5-mile per hour difference, at least on the turbine building siding, there are very distinct design criteria that seem to be imposed in their design.

And that starts to get a bit troubling. So, anyway, I think we've raised the concern, and I do, I appreciate the feedback because we now do have on the record that the guidance that you used and the basis that you used for selecting those two different wind speeds.

So we'll toss it back to the staff to get back to us at some time. This by way, it is obviously a generic issue, it is not particularly a US-APWR issue because we do understand where you selected your values from for US-APWR design.

However, it may have implications on US-APWR design if, indeed, the design criteria for certain structures are substantially different based on those two different wind speeds.

And, again, I mentioned the turbine building because I at least saw something there. So, thank you.

MR. NAGAI: Okay. So, next one, Section 3.5, there was a question about technical justification for determining probabilities of internally generated

missiles and there's a DCD statement that the probability of internal missile generation is maintained with ten to the minus seven and the Chairman was asking the basis for that.

And, also, you pointed out inconsistency between that probability and the one used in Chapter 19, and Ogasawara-san has an answer for that question.

MR. OGASAWARA: Yes. This is Hikaru Ogasawara again. It is described in DCD 3.5.1.1.1.1 --

(Laughter)

(Simultaneous speaking)

CHAIRMAN STETKAR: 3.5.1 --

MR. OGASAWARA: 1.1.1.1, four times one.

CHAIRMAN STETKAR: Four ones, 3.5.1.1.1.1, okay.

MR. OGASAWARA: The probability of occurrence due to the potential missile from piping is less than ten to the minus seven after that. If piping is evaluated in Section 3.6 were to rupture, the piping is held by its support.

However, the probability of occurrence, P1, remains less than ten to the minus seven since the section remains attached to the remainder of the piping system.

This sentence means that the piping break is considered according to Section 3.6 and possibility

of damage by missile caused by piping break is to be ignored, that is to say possibly less than ten to the minus seven.

CHAIRMAN STETKAR: I guess, you know, this is another situation where, I understand that argument by the way, that regardless of what the piping failure frequency is, the frequency of missile, the conditional probability of missiles being generated from a pipe break, you know, would be lower than that frequency because of the supports and the anchors and things like that.

The basic concern that I'm raising is not so much about each individual piece of the evaluations that are presented in DCD Chapter 3, it's the fact that statements are made that probabilities are less than ten to the minus seven, which is a very small number, without any evidence of actual supporting analyses to justify those small numbers.

And the question is, if you're going to draw a conclusion that is based on a small number you should be able to show the evaluation that was done to justify that small number.

So, for example, in a pipe break, if the pipe break frequency was ten to the minus three per year, why are you confident that the conditional probability of a

missile is a factor of one in 10,000? Follow me?

MR. OGASAWARA: Okay.

CHAIRMAN STETKAR: Yes, and that requires some analysis. It doesn't require what we just think it would be a lot smaller, because a lot smaller could be one in 1000, which is pretty small, but it's still then a factor of ten times higher than ten to the minus seven.

So that's what I'm basically challenging, is that if you're drawing these conclusions based on numerical arguments, like a frequency of ten to the minus seven per year, I'm asking whether you have the analyses to provide confidence that those very small numbers are justified.

And it's for a variety, I mean, I brought up the examples of the LOCA frequencies because they're the easiest things to point to, but it also applies for several of the other types of missiles that you address from rotating equipment and load drops and things like that.

MR. TAKAYAMA: Is it okay?

CHAIRMAN STETKAR: Sure. Well, I mean, that's, I'm trying to raise the concern here a little bit because I hear what you're saying on pipe supports qualitatively, but I still don't hear the fact that you've looked at those piping sections and can justify

that the conditional probability of a missile being ejected is less than ten to the minus three, or ten to the minus four, or whatever it would take if you look at the pipe break frequency.

In a similar manner to, you know, you did do the turbine missile analysis, that essentially looks at all of those conditional probabilities so there is some basis for saying yes, I have confidence that the frequency of a turbine missile ejection is less than whatever it is, ten to the minus five per year.

MR. TAKAYAMA: My name is Yoshihiro Takayama. We perform pipe break protection design in accordance with Standard Plan 3.6.1, 3.6.2, and other practical place position 3.3 and 3.4.

So our design is -- in comparison with Standard Review Plan acceptance criteria, so practical risk, concept of a pipe break is relatively small.

We use deterministic design for pipe break. Then we, because that physical damage by pipe break missile is relatively small, which is we think ten to the minus seven.

CHAIRMAN STETKAR: See, the last statement there that I heard is you say "we think ten to the minus seven," and I'm asking for what's the justification for why you think it's less than ten to the minus seven

because ten to the minus seven is a very, very small number.

MR. TAKAYAMA: But we --

CHAIRMAN STETKAR: If you think it might be less than ten to the minus six, which is also a very, very small number, that's still a factor of ten times higher than ten to the minus seven.

So I'm asking you why you think it's less than ten to the minus seven and not just less than ten to the minus six? Follow me? I mean it's, what I'm really raising, and perhaps I'll ask the staff about this because the staff has accepted what you've developed.

They didn't have any, I didn't see any open items or lingering questions about essentially that screening process that you went through for all of those different potential missile hazards.

I'll ask the staff then how the staff has confidence that indeed those ten to the minus seven numbers that are cited in the DCD are reasonably acceptable.

MR. GALVIN: We didn't bring the staff today. We'll have to get back to you.

CHAIRMAN STETKAR: Yes. No, that's -- so we'll just leave that open, but I'll leave you off the hook for now because I do, again, I appreciate you've come

back and told us at the references that you used in terms of the Standard Review Plan, the basis that you used in terms of the piping, in terms of supports and hangers and things like that, to give you confidence that the conditional probability of missiles would be small.

What I'm questioning is the justification for smaller than that very small number. So I'll ask the staff then because they've accepted those values to see if they can come back to us at sometime in the future and provide us a little bit better confidence about why they feel those numbers are reasonable.

MEMBER SCHULTZ: Again, I will just comment that I appreciate that you framed the issue well, because this is what we were describing and discussing yesterday.

In the past, before anyone felt obligated to assign a number to the conclusion of the discussion, negligibly small, would have been an appropriate conclusion related to this.

But if it's felt, if we feel a need to then assign a number and we choose to assign a number of ten to the minus seven to everything that we believe is negligibly small, we document that number, and the concern would be that sometime someone would use that number and it really doesn't necessarily represent reality.

It represents negligibly small, but it can be --

CHAIRMAN STETKAR: It's a good --

MEMBER SCHULTZ: -- a concern if it's in fact then used later on as a real value in an analysis. It's happened before in other industries. We don't want it to happen here.

CHAIRMAN STETKAR: That's a bit of the problem. Once, as Steve said, once you've put that number, especially in the Design Certification Document --

MR. TAKAYAMA: It will --

CHAIRMAN STETKAR: -- you've essentially raised the bar, you've leant credibility to that value and if you don't have the supporting analyses to justify that it could be dangerous in many different ways.

MR. TAKAYAMA: I understand.

CHAIRMAN STETKAR: Not dangerous to safety, but dangerous in terms of misinterpretation of what that number might mean or the intent of the conclusion, the engineering conclusion.

So I'll ask the staff to come back to us sometime in the future. We'll leave that as an item that we'd like the staff to get back to us, okay.

MR. GALVIN: We'll track with the

transcript.

CHAIRMAN STETKAR: Yes, thanks, Dennis. Thank you.

MR. NAGAI: So next question. While we were discussing Section 3.5 the Chairman pointed out that the analysis performed for MUAP-07029, it appeared that the failure of hydraulic system and over-speed protection system, in other words the entire system, was not considered in the analysis.

And Minami-san of MHI would like to provide additional information regarding that question.

MR. MINAMI: This is Minami speaking. And before starting our discussion I would like to give a brief explanation on our control and protection system and for better understanding your comment yesterday.

And our system, this picture shows our control system and contains main stop valve, main control valve, RSV and IV. And high pressure. High pressure oil is supplied. High pressure oil is supplied to each of the valves through the high-pressure pump, two high-pressure pumps.

And the high-pressure pump is supplied to the servo valve, except for the RSV. Because RSV is operated in on/off most so no servos. And the high-pressure is supplied to the servo valves and the

regulated pressure oil is supplied to the piston, which is connected to the valve directly.

And each valve is equipped with dump valve. And the dump valve is connected to emergency trip headers. We have two emergency trip headers. One is for MSR, MTSD and RSV and the other is for MPCV and IV.

And if the pressure in the emergency trip header is reduced the dump valve is opened and the high-pressure oil in the piston is drained to the drain line. And the valve closed.

And high-pressure emergency trip header is connected to -- one emergency trip header is connected to OPC, over-speed protection control. And the other header is connected to EOST, which is being consist of four solenoid valve establishing one out of two, twice project, right?

CHAIRMAN STETKAR: Yes.

MR. MINAMI: And this, at the bottom, on the right side, you will see a spill pickup, right? We have independent spill pick-up sets. One is for control. One set is for control and the one set is for protection system. Independent system.

And we have three pick-ups for each sets. And two out of three signal is sent to the solenoid valve through turbine protection system and the safety logic

system.

And the OPC has two solenoid valves. And if one solenoid valve opened the -- The two solenoid valves.

CHAIRMAN STETKAR: Yes, 52A and 52B. Are the valve numbers, you can't see them on here.

MR. MINAMI: Okay. And at least one SV must be open for tripping. Not tripping. Not tripping.

CHAIRMAN STETKAR: You need both of them to open for tripping.

MR. MINAMI: Right.

CHAIRMAN STETKAR: Yes.

MR. MINAMI: And USV has four solenoid valve and one have to trip twice for --

CHAIRMAN STETKAR: Right. Those valves numbers are, I hope, 51C, D, E and F. So far my equations are correct. Go on.

MR. MINAMI: In addition to EOST we have MOST, mechanical over-speed trip devices, which is mainly operated by mechanical parts and are completely independent from the EOST. Speed sensor is also independent from --

CHAIRMAN STETKAR: Yes.

MR. MINAMI: And so this is the outline of our hydraulic control and protection system. And, if my

understanding is correct your comment you give us yesterday is, in the MUAP in our technical report we are addressing the left side, mainly left side, which includes a main valve. We are addressing only left side.

And we should address, what we should focus on, the right side equipment or component. We need to address right side component.

CHAIRMAN STETKAR: That is exactly correct. That's my concern. Well I don't want to say focus, I want to say the entire frequency of over-speed failure should include everything that's on this drawing. And that it's not, as you explained the system, as is my understanding of the system, it's also not as simple as a line for the left and the right because of the way that the over-speed trip controls operate the solenoid valves.

So for example, as you mentioned one pair of solenoid valves, the 52 number solenoid valves, affect one set of main valves. And the 51 set, the one-out-of-two taken twice, affect a different set of main valves.

So you cannot just evaluate the solenoid valves in isolation because different combinations of failures of the solenoid valves on the right side will effect different combinations of the main valves on the

left side.

So the only way to really evaluate the entire failure frequency is to look at everything that's on the slide in front of us together. And that was basically my comment.

And you're exactly right, the main concern that I have is that the current analysis in MUAP-07029 addresses only what you've numbered 1, 2, 3, 4 on the left side of what we're seeing on this drawing.

Now how important is the equipment on the right side when you put everything together? That I don't have a clear answer for.

MR. MINAMI: Okay. Next slide please. And I'd like to move to the issue we need to discuss today. And last night I checked our back data for the technical report of 07029 and I confirmed the data, the data is taken from all the Japanese PWR units, more than 20 units.

And the data is the data as of 2007, because this report is write in 2007. And I confirmed that such events as below counted as valve or system failure. Such incidence includes servo valve malfunction due to degraded oil. Servo valve or solenoid valve malfunction due to inadequate maintenance or single failure of the components such as LVDT.

So in this part I want to say that even

single component failure occurred, I counted it's one. One event.

CHAIRMAN STETKAR: Are these, the failures that you're discussing, failures related to the valves on the left of the drawing? The turbine control valves, stop valves and intermediate valves and reheat stock valves. Did you look for failures of the solenoid valves or the equipment on the right-hand side of your drawing?

MR. MINAMI: Yes.

CHAIRMAN STETKAR: You did?

MR. MINAMI: Yes.

CHAIRMAN STETKAR: And you found none? Or you found one of those?

MR. MINAMI: One or two --

CHAIRMAN STETKAR: One or two?

MR. MINAMI: Yes. For example, I found some solenoid valve malfunction due to, I think, due to inadequate maintenance. But --

CHAIRMAN STETKAR: But what you're saying is that your database does include the equipment on the right-hand side of that drawing, the solenoid valves?

MR. MINAMI: Right. And as we discussed yesterday, single malfunction do not lead to the turbine fail. But I assumed single malfunction is equivalent to the turbine trip.

CHAIRMAN STETKAR: Could, yes.

MR. MINAMI: So and the third sentence is no actual unexpected turbine over-speed by instant observed during investigation period. Strictly speaking I found one turbine over-speed event. But it was due to misoperation by the operator. So I didn't include such events.

So if I exclude such data I didn't find any unexpected turbine over-speed actually. So my conclusion is, like that, all the instance including any single failure of the component within the entire system are counted as MTCV or protection system failure for conservative purpose.

So I agree that my event categorization is misunderstanding or confusing. But what I did is I think nothing can be --

CHAIRMAN STETKAR: Well I see at a high level what you have done. And again I'll come back to saying that when you look at -- First of all the failures of any of these pieces of equipment are very small.

If I take the failure rate model that was used in MUAP-07029, in the MUAP, for the valves on the left, the main valves. If I translate that annual failure rate into a per-demand, in other words, how likely is it that the valve fails to close, those failure

rates are on the order of about, the low end, the ones for which there were no failures observed, are about 2.6E to the minus 4.

In other words, about three failures in 10,000 demands. For the high end, where there were some valve failures observed, they're about 7E to the minus 4. About seven failures in 10,000 demands, less than one in 1,000.

So they're still very, very reliable valves. Your data that you've compiled or looked at for the solenoid valves in the hydraulic system confirmed that they're also very reliable valves. My concern is that the integrated model, the combined model of everything on the slide before, combines a large -- Well I have to be careful.

A number of combinations of two failures and three failures together, not single failures, but two and three failures together. But a large number of those combinations that add up. In other words, you know, ten 10 to the minus six combinations gives you one E to the minus 5.

And what I'm interested in understanding is how do the combinations of, in some cases it's two failures some cases it's three failures, for the equipment on the right side of the slide when it's

combined with the equipment on the left side of the slide, how do those missing combination affect the conclusion that the test interval will still give us less than 1E to the minus 5 per year?

And I think it might, but I don't base my conclusions on what I think. I base my conclusions on what I can look at and see what the analysis has done. So this is encouraging information to say that we have not had any hardware related over-speed incidents. And that you have not identified any failure data that would lead us to believe that the solenoid valves in the hydraulic system had very high failure rates, which is good.

You know, if you found many of those failures that would be a potential real problem. So it's encouraging that the failure rates of those valves are low. And given the data that you have for the main valves there's some evidence, from what you looked at in your data, that the solenoid valve failure rates may be comparable to those main valves.

In other words, two to five to seven, E to the minus 4 per demand, that they're not ten to the minus 2 and they're not ten to the minus 7. But that still doesn't give us the confidence that the integrated model would still support the 1E to the minus 5.

It gives me additional confidence. But it doesn't give us that confidence that the full model would support that.

MR. MINAMI: And I think we need to consider the fact that we have, our system has the capability to test solenoid valve online. And I'm not sure our customer is really doing such kind of --

CHAIRMAN STETKAR: That is, by the way, that's always a question, because the way the models are developed -- And I know we're going into a lot of detail here, but we will finish by noon. The way the models are developed that that testing interval and whether it's really used by your customers to support the number of failures is very, very important.

Because the failure probability and the way the models are developed depends directly on those test intervals. If your customers are only testing the solenoid valves once per year that failure data accounts for that once per year testing. If they're actually testing them once per month the failure data accounts for the once per month testing.

So it's important to understand what that customer base is telling us in terms of the failure data and the types of testing that the customers are doing to confirm those failure rates. That's in some sense a

related but different issue in terms of the completeness of this evaluation.

MR. MINAMI: But if we consider such factor as we can be tested, you know to carry online, I think our result, our conclusion in this technical report is not reasonable I think. So --

CHAIRMAN STETKAR: Well we're getting into a lot of detail. We can discuss this a little bit more perhaps after the meeting. Because we probably have some other items to discuss this morning and we do need to end by noontime.

It's good, I really appreciate the work that you did to look for that data. And I will tell you from my perspective it's encouraging because you didn't identify anything that would lead me to have a definite concern. It's still I'm not completely satisfied with the completeness of the evaluation to justify that small number. Okay?

And as I said, if you want to we can discuss a little bit after the meeting and I can show you a little bit some more of my concerns. And, again, I'm not concerned in an absolute sense of whether or not the system is safe. No single failure will cause failure of the system. The data that you presented do not give me concerns about the reliability of the individual valves.

It's just having confidence that that 1E to the minus 5 number is supported by a complete evaluation. And it's especially important not only for the design certification it's important for the COL applicant because they're relying on that value for confidence that their unfavorable design configuration at Comanche Peak is still acceptable. So that 1E to the minus 5 is even more important for them.

Thank you, by the way.

MR. NAGAI: Okay. Next question was also related to this technical report. How MHI determined the failure rate of stop and control valves since zero events exist to evaluate against. So Minami-san.

MR. MINAMI: Yes. On this picture it shows how we estimated the failure rate in case of no events. And the black solid mark on the left side is the starting point of evaluation. And the left side black solid mark is the timing of investigation. And between these two black marks when we didn't, no incident, we assume, incident, just after incident, just after investigation.

And including this assumed event we calculated the failure rate. I think this way gives us some conservative number.

CHAIRMAN STETKAR: Yes, okay. Thanks, that actually does answer my question that I had. I

thought this is what you did, but it wasn't completely clear because in the report you talk about mean values of zero and 95 percent confidence values of some number. And then you use the 95 percent confidence value, but really the value that you're using was calculated this way.

MR. MINAMI: Yes.

CHAIRMAN STETKAR: For those cases where you had zero failures?

MR. MINAMI: Yes.

CHAIRMAN STETKAR: Okay. Thank you. Let me make a note on that. That one is done.

MEMBER BROWN: This still the same topic? Figured as much.

CHAIRMAN STETKAR: You have a question?

MEMBER BROWN: Not on -- Well you triggered my thoughts yesterday and so I did some review yesterday, because we did Chapter 10 how long ago? A while ago, on the over-speed trip system. And I had gone through it then and didn't have a big problem.

But I've learned subsequently about another point. Now you all have a mechanical over-speed trip and I used to operate machines that had mechanical over-speed trips and that's all they had. There was no other, I'm going back 35/40 years.

And we didn't allow electronic trips at that time, just because of the reliability of them. This is for naval vessels, war ships. And so we periodically tested those, but it wasn't frequent. It might be at an overhaul or a maintenance period, a major maitnenance period, stuff like that. So we're talking multiple years in between testing those.

And I can't remember in my experience with the naval nuclear program ever seeing -- The mechanical over-speed trips always worked. They may not be quite as fine tuned, you can say well they're going to trip at 105 percent or 110 percent, might be 112, might be 108. But not as crisp as the electronic systems are.

So when I looked at the systems I see a very clear independence between your electronic over-speed trip and your mechanical over-speed trip. And this is an information question from the standpoint, how often -- I went back and looked it up. I couldn't find it real quickly because I just lost track of how to do it. How often do you all test the mechanical over-speed trip in your all's, is it covered under the ITAAC in your DCD?

MR. MINAMI: No, I don't think so.

MEMBER BROWN: Is it every refueling outage? Or is it every ten years or anything like that? I mean does anybody have a feel for that?

MR. MINAMI: We usually recommend two kind of test. Two kind of test. And the first test is actually over-speed, actually over-speed --

MEMBER BROWN: Yes, you raise the speed of the turbine up to a certain point and get the centrifugal switch to operate and that triggers the whole rest of the mechanical chain.

MR. MINAMI: Right. And this kind of test is carried out just before the shut down for refueling.

MEMBER BROWN: Okay. So every refueling?

MR. MINAMI: Right.

MEMBER BROWN: So about every two to three years?

MR. MINAMI: Two years or three years, right.

MEMBER BROWN: Okay.

MR. MINAMI: And in addition to the first test we usually recommend to do the oil pressure test. Just supply the high-pressure oil to the eccentric weight. And the high-pressure will high-pressure, overcome the springboard and that eccentric weight --

MEMBER BROWN: All right. You've said enough. I understand that.

CHAIRMAN STETKAR: How frequently do you do the oil test?

MEMBER BROWN: Well he said, if I got it, he does it before they do the over-speed test.

CHAIRMAN STETKAR: Right before. I missed that part.

MEMBER BROWN: I thought that's what you said, is that correct?

MR. MINAMI: No, that test is once a month.

CHAIRMAN STETKAR: Once per month for the oil test.

MR. MINAMI: Right.

MEMBER BROWN: Okay. That would tear my hair out, but you couldn't do --

CHAIRMAN STETKAR: We used to do it once per month and we only tripped about once every four times, so.

MEMBER BROWN: That's a problem.

CHAIRMAN STETKAR: It was.

MEMBER BROWN: Okay. But I haven't finished quite yet, so you're not off the hook yet. Now, what if you run that test on the mechanical over-speed trip and it fails? Do you have to fix it before you go back into operation? Are you allowed to restart and operate through another operating period of two to three years with and inoperative mechanical over-speed trip function?

MR. MINAMI: I'm sorry, I can't answer. I don't have answer your question at this moment. So I have to take my time before --

MEMBER BROWN: Okay. I'm asking because if the answer is yes then that raises some questions, which I did not look at, on the EOST. Your diagram, if you read all the words in the DCD, it talks about, you know, it's got independent sensors from the TPS for the control system side.

If you look in your diagram you've got separate sensors for it. And it goes through separate electronics, separate processing units. At least that's the appearance, even though it's housed in the same cabinet as the EOST is.

But then if you say it's okay to operate with the mechanical system to of service that raises questions about the EOST and how those sensors are fed into the processing part.

As an example, in other words, you want now -- It's two out of three, I think, if I remember, it might be two out of four. It's --

CHAIRMAN STETKAR: Two out of three.

MEMBER BROWN: Two out of three, right? For tripping. And you then want those three little channels to be independent of each other. In other

words, independent power supplies, a sensor feeding each of the channels. That would make everything, you could put a wall between them, and everything would be totally independent. And that's independent of your control system.

If the sensors feed, all three, feed each of the three channels of processing and you only have say two power supplies feeding all three, you know, optioneered so you have continuity of power, all three of the processings, you've now destroyed your independence of those three channels.

And on a software-based system it's even worse than on an analog-based system in terms of having something fool the system and then potentially result in inoperation, other than in between your checks, which you may find something for some period of time. None of which is described in your DCD in any detail as to how that independence would be maintained.

So that's the purpose of my question. It was raised, I've got a similar issue, or a similar situation, that I'm evaluating for another project, trying to get some definition. And then doing some comparison on another one.

So that's why I bring the point up. But you don't have an answer of whether you're allowed to go back

into operation with the mechanical over-speed trip out of service, for two years say?

MR. MINAMI: But from design viewpoint, we also recommend to do the online test of EOST.

MEMBER BROWN: Oh no, I understand all that. But that's not, the online testing is different. That's the testing for a specific failure, a sensor fails, a processor fails, a power supply fails. With software-based systems you can have noise come in, I've seen it happen that's why I'm asking, and it literally corrupts the data and it disable the over-speed trip. In other words it just never sees that. You can over-speed and it just doesn't know it's going to over-speed.

MR. MINAMI: What I want to say is that during the EOST online test the EOST doesn't work.

MEMBER BROWN: No, I understand that.

MR. MINAMI: Okay. But if MOST is working, if system separation occurred MOST would --

MEMBER BROWN: I'm not worried, as long as the MOST is not operational I don't have a problem. My only concern is you come out, you've tested the over-speed function coming out of the refueling -- Going into the refuel --

CHAIRMAN STETKAR: Charlie, we do need to finish the meeting. Let me go back. If you had an

integrated model for everything that was shown on the first slide you could address these questions. Because the integrated model would look at the probability of failure of the mechanical over-speed trip.

It would look at the probability that it's unavailable due to maintenance. It would look at the probability of failure of the speed sensors. It would look at the actual logic for the speed sensors. It would look at common cause failures. You don't have that model so you can't actually answer the types of questions or determine how they might affect the reliability.

And Charlie's raising concerns from kind of a deterministic sense without the notion of how important they might be. And you can't answer questions about important they might be because you don't have the analytical model developed to sort of explore those questions.

You know, how important is it if the mechanical over-speed trip is out of service for a whole year due to, you know, inoperable. And if the sensors indeed do have some type of common corruption. If you have a model you can --

MEMBER BROWN: And corruption is the key word. I don't care what made them bad. You can't answer -- It's what they do to the processing system if it's a

software-based system. That's the issue. You can lock up the software-based decision maker and not have a trip. And that's different, I understand it's a different type of failure, that's all, that people have not thought of.

CHAIRMAN STETKAR: My whole point is that an integrated model could help to answer several of these questions. And you don't have that.

MEMBER BROWN: No, the simple answer is if you always repair it before you go back into operation. You know it's working when you go in, you don't test it again until the end.

So if it's not going to work in between you're not going to figure that out unless it fails in a manner to trip the turbine, which is a failure mode. Then you will know it. Then you may want to go back up while you're replacing a valve or solenoids or what have you and then the next question comes into play.

So that's the point. We can get on. There wasn't enough definition in what we have in terms of the design. I've got that in one other project and now that has raised other questions because I didn't understand the discussion relative to the electronic functions.

I turn it back to you. I'm sorry, John. I wanted to get that --

CHAIRMAN STETKAR: No, it's okay. No,

they're good questions. They are good questions. They can be evaluated if you look at the whole system in total.

MR. MINAMI: I know.

CHAIRMAN STETKAR: I'll keep coming back to that, that you can answer many of these questions if you have that whole model. Or you could at least explore, you could do sensitivity studies, for example, to explore some of the implications of the types of issues that Charlie is raising.

MR. MINAMI: Okay.

MR. NAGAI: Okay, next moving on to Section 3.4, there were four open items. The first one was related to a break in charging pump piping on the west side of the building. The question was, "What is the justification for the flooding level being limited since at switch over should occur for the charging pump to draw suction from RWSP?"

CHAIRMAN STETKAR: Right.

MR. NAGAI: And Nishio-san from MHI would like to provide an answer to that question.

MR. NISHIO: This is Hiroki Nishio again. And MHI has already submitted this RAI response related to this issues. RAI Number is 841-6055.

CHAIRMAN STETKAR: 841-6055?

MR. NISHIO: Yes.

CHAIRMAN STETKAR: Okay.

MR. NISHIO: Yes. And then -- that when charging line failure and then letdown isolation valve will close. Ater that, delivery of the VCT going down and no alarm will be activated. And then after that the water source of the suction line will be switched to the VCT to the RSWP, or the multi-line.

So the charging pump will continue to run but that based on the VCT alarm, operator can stop the charging pump. According to the flooding operation, we counted that a maximum of 15,000 cubic feet is taken into account, or taken into accounted into the flooding analysis. The other hand, the maximum flow rate of the charging pump -- it means flow over the charging pump, is about 100 GPM. Oh let's see 440 GPM of flow, 100 cubic.

So then we have enough time to reprogram -- or whatnot. We have time, a little of time to stop the charging pump.

CHAIRMAN STETKAR: Okay. Thank you for pointing us to the RAI, we'll look. We'll get a copy of that RAI and the response. I understand what you're doing. You're basically taking credit for the operator responding to the alarm and tripping the pump.

MR. NISHIO: Yes.

CHAIRMAN STETKAR: So we'll take a look at that. Thank you.

MR. NAGAI: Okay. Next one was this time east side of the reactor building a non-radiological controlled area, the Chairman was asking what happens to the result of the evaluation if the door from the area to process building somehow disturbs the flow from those two buildings. And Nishio-san, again, has an answer to that question.

MR. NISHIO: Yes. Let's see. In the, you know, the flooding analysis we consider that one area, both sides of the reactor building and process building too.

CHAIRMAN STETKAR: Yes.

MR. NISHIO: And because both buildings were separated by the fire work door, watertight door. But we have assumed that some, maybe it is not register, it doesn't register to the worker so we are -- We consider the worker in the reactor building going to the process building and moreover we have the drain line, a drain line from the process building to the reactor building. So that mean the bypass line, there becoming the bypass line. So the level of the reactor building and the process building will be ultimately the same.

CHAIRMAN STETKAR: Okay. Yes. And I know

that's what you did. The drain lines do communicate directly? I mean even if the door was absolutely watertight you would still have propagation through the drain lines?

MR. NISHIO: Yes, the --

CHAIRMAN STETKAR: Or do the drain lines in those areas have check valves in them?

MR. NISHIO: Sometimes. Sometimes in located in the RB. And I know sometimes in the process building. Just during process building go to the reactor building.

CHAIRMAN STETKAR: So you go from power resource to the reactor building but not the other way?

MR. NISHIO: No, there is no check valve.

CHAIRMAN STETKAR: Okay.

MR. NISHIO: So there is a pipe restraint.

CHAIRMAN STETKAR: Okay. I'd have to -- I think in the interest of time, that's starting to get a little more complicated than I can understand. I'm actually more concerned in the reactor building and the power source building of floods from the service water than I am from that door. I think I understand what you did and why you did it. But I need to think about it a little bit more.

MR. NAGAI: Okay. We have two more. One

is also associated with Section 2.4. The Chairman pointed out two potential additional water sources, such as ESW, CCW heat exchangers. And Nishio-san has answer to that question.

MR. NISHIO: If the -- either repeat occurrence due to the human error or -- this, flood water is collected through the drain line system and to the non-radioactive reactor building sump located in the lowest level in the NRCA, non-radioactive control area.

And so the RB sump, reactor building sump, has a high water level and the operator can know that a leak, there's a leak in the reactor building. I mean, that's in this side. And so reactor can take action to some containment measure so when there is -- so but this is not, this kind of event is just operating issue. So in the flooding analysis we put in there some bounding value to the analysis and we evaluated. So is it --

CHAIRMAN STETKAR: Do you actually have a flooding analysis that looks, I mean that specifically addresses the situation that you were describing? In other words that addresses the capacity of the drain line, the timing of the alarms and the timing of the operator action to isolate the flood hazard for all of the potential flood hazards in each area?

I mean, I hear what you're saying about what

could be done. I'm asking you whether you've actually evaluated those types of scenarios for different flooding sources.

MR. NISHIO: I can't respond right now. So okay. Okay.

CHAIRMAN STETKAR: I'd appreciate, if you have that, I'd appreciate seeing it because I'd like to understand a little bit better what you did. Thank you.

MR. NAGAI: The last one for the main steam feed water piping area of the reactor building at 76 feet and 5 inches. Question was if a break occurs in that area can the doors to that room withstand a likely pressure transient caused by the break.

MR. NISHIO: First in this FW view are closed compartment but with the blow off panel, the rupture panel, is installed.

CHAIRMAN STETKAR: Oh, okay.

MR. NISHIO: And then so if that room pressure goes high the blowout panel rupture and the pressure will go to outside.

CHAIRMAN STETKAR: I missed the blowout, for some reason I either didn't remember or I missed the blowout panels. So thanks. That answers that one. Thank you.

MR. GALVIN: John, we do have Mohamed

Shams, the construction engineering branch chief who could address, at least speak to your concerns on severe wind versus extreme wind. I don't know if you want to do that now or after you close.

MR. SHAMS: I could take one minute to explain that.

CHAIRMAN STETKAR: Sure, let me make sure, have you covered everything?

MR. NAGAI: Yes.

CHAIRMAN STETKAR: Okay. Great. Thank you, very much. I really, by the way, I appreciate -- One of the reasons why we schedule this for two days because it does give you the opportunity to get back to us. And I really appreciate the effort getting back to us. And you've clarified several of the issues. You've answered a few of them, so I do appreciate that.

MR. NAGAI: Thank you, Chairman.

CHAIRMAN STETKAR: The members have any more questions for MHI? I know, Charlie, you raised a couple of questions.

MEMBER BROWN: That was the last thing. Somehow I've got to think about that. Try to get some more info on it.

CHAIRMAN STETKAR: So yes, looks -

MEMBER BROWN: -- be helpful on that,

eventually.

MR. SHAMS: So my understanding is that the Committee asked about the --

CHAIRMAN STETKAR: Just make sure we have your name for the record.

MR. SHAMS: Sure. My name is Mohamed Shams, S-H-A-M-S.

CHAIRMAN STETKAR: Thanks.

MR. SHAMS: And I'm the branch chief for the Structure Engineering Branch.

CHAIRMAN STETKAR: Thanks, Mohamed.

MR. SHAMS: So my understanding of the question is the Committee looked at the values reported for the severe wind, the speeds reported for the severe wind versus the hurricane. And one being 155 miles per hour for the severe wind versus the hurricane being 160. And the probabilities of exceedance for the two was not proportional to the modest increase in speed.

The answer to that is because they are not coincident, those two valleys don't happen at the same location. 155 mile per hour severe wind corresponds to more like 290 mile per hour hurricane that has the 10 to minus 7 probability of being exceeded.

On the other hand 160 mile per hour hurricane at 10 to the minus 7 corresponds more like to

a 90 mile per hour severe wind. So what that means ultimately is that MHI selected a severe wind value that essentially envelopes the entire continent of the United States. But the hurricane wind leaves portions that are not enveloped.

That's addressed later on in a COLA application whether or not that site will be appropriately covered by the parameters. And we would take exceptions at that time.

MEMBER RAY: The difference is what causes the wind. Whether it's a straight-line or a hurricane.

MR. SHAMS: Right, yes.

CHAIRMAN STETKAR: Let me phrase it differently. What is the 10 to the minus 7 straight-line wind speed? You said something like 200 and some odd miles an hour?

MR. SHAMS: 209.

CHAIRMAN STETKAR: 209, okay. I'll have to think about this, because it's still --

MR. SHAMS: So it's for the site versus for the --

CHAIRMAN STETKAR: Well I understand the site because I actually looked at the site data. I know where they came up with the 94 miles per hour. I know where they came up with their hurricane for the site.

What still is confusing to me is that we're characterizing a 155 mile per hour wind as a once in 100 year. If I operate the site for 100 years I would expect to see it once. We're characterizing a 160 mile per hour wind as occurring once in 10 million years, that if I operate the site for one million years I'd only have a ten percent chance of seeing a five mile per hour higher wind speed. And that to me still doesn't make sense.

MR. SHAMS: Yes, maybe I'll try to explain better. As a COLA we select a site, these numbers wouldn't be those numbers. You know, if a COLA is to say select a site that would have the 160 mile per hour 10 to the minus 7, they would have a 90 severe wind that would not be exceeded more than one time in 100 years.

CHAIRMAN STETKAR: Right, got that. But the bigger concern is not so much the numbers roulette that we're playing here. The bigger concern is what implications do those two values have on making design decisions for the plant? What differences in the design arise because of something that's assigned 155 mile per hour wind speed at a nominal occurrence interval of once in 100 years.

And what different design decisions are made relating to that 160 mile per hour wind speed, or lack of design decisions. I don't need to make a design

decision for 161 miles per hour because by definition it is less than 10 to the minus 7 per year.

MR. SHAMS: Sure.

CHAIRMAN STETKAR: And yet I expect a 155 mile per year to occur once every 100 years. So am I either not including something in the design that I ought to, because I'm just below that magic 10 to the minus 7 number at 160.1 miles per hour?

Whereas, I'm designing for 155 mile per hour wind in once in 100 years. So am I losing something, or am I including something in the design that I ought not to because I'm designing to this 155 mile per hour wind when I don't really expect it to occur?

MR. SHAMS: Yes. I mean to best answer that --

CHAIRMAN STETKAR: And both of those questions have implications.

MR. SHAMS: To best answer that is two values are going to be treated differently in load combinations. One, that being the severe wind, it gets treated in load combinations that have lower, if you would, factors of safety on them. The one with the higher probability would have a load factor of like 1.7 so we would double that.

To answer your question, am I losing

something by considering just the 160 as opposed to, you know, 290? Yes. As it is you wouldn't be able to --

CHAIRMAN STETKAR: But the way to think of it is the 160 versus 290. Not the 160 versus 155.

MR. SHAMS: Yes.

CHAIRMAN STETKAR: Okay. That helps a little bit. That does help.

MR. SHAMS: And, you know, by virtue of that is the design as it is wouldn't be suited directly to like a South Florida site. You know, we --

CHAIRMAN STETKAR: Oh sure. Yes, that's clear. Yes, okay. Thanks. I'll have to think a little bit more about that. But that does help, thank you.

Anything else from the staff?

MR. GALVIN: That's all we have right now.

CHAIRMAN STETKAR: Okay. Anything else from any of the members? If not adjourned. Thank you very much for everyone, staff and MHI Lumina. Thanks for all the feedback on the questions. I think we got a lot of things clarified. Anything else from anyone in the room? If not we are adjourned. Thank you.

(Whereupon, the meeting in the above-entitled matter was concluded at 12:06 p.m.)

Presentation to ACRS Subcommittee

Chapter 3:
Design of Structures, Systems, Components and Equipment (Except 3.7/3.8)

November 20, 2013

Mitsubishi Heavy Industries, Ltd.

Presenters

Thomas Hicks – Lead Presenter (Sections 3.1, 3.2, and 3.11)

Mark Biery – Lead Presenter (Sections 3.3, 3.4, 3.5, 3.6, 3.9, 3.10, 3.12, and 3.13)

Hiroki Nishio – Technical Expert

Richard Samples – Technical Expert

Yukihiro Nishizawa – Technical Expert

Hikaru Ogasawara – Technical Expert

Yoshihiro Takayama – Technical Expert

Yoshihiro Minami – Technical Expert

Msashi Ito – Technical Expert

Atsushi Matsumoto – Technical Expert

Masatoshi Nagai – Licensing

Contents of DCD Chapter 3 (1/3)

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Section No.	Description
3.10	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment
3.11	Environmental Qualification of Mechanical and Electrical Equipment
3.12	Piping Design Review
3.13	Threaded Fasteners (ASME Code Class 1, 2, and 3)

Section 3.1: Conformance with NRC General Design Criteria

3.1 Conformance with NRC General Design Criteria

Subsection No.	Description
3.1.1	Overall Requirements
3.1.2	Protection by Multiple Fission Product Barriers
3.1.3	Protection and Reactivity Control Systems
3.1.4	Fluid Systems
3.1.5	Reactor Containment
3.1.6	Fuel and Reactivity Control
3.1.7	Combined License Information

3.1 Conformance with NRC General Design Criteria

- US-APWR conforms with applicable criteria in 10 CFR 50, Appendix A, the General Design Criteria (GDC).
- Section 3.1 describes how the US-APWR design meets the applicable GDC and refers to specific DCD sections for further technical information.

3.1 Conformance with NRC General Design Criteria

- **Major RAIs (Open Items)**
  - ✓ There are no Open Items in Section 3.1.

Section 3.2:
Classification of Structures, Systems, and Components

3.2 Classification of Structures, Systems, and Components

Subsection No.	Description
3.2.1	Seismic Classification
3.2.2	System Quality Group Classification
3.2.3	Combined License Information

- **The US-APWR SSCs are classified according to nuclear safety classification, seismic category, quality groups, quality assurance classification, and codes and standards.**
  - ✓ Seismic classification is addressed in Subsection 3.2.1.
  - ✓ Equipment classification is addressed in Subsection 3.2.2.

- **There is one Open Item in Section 3.2.**

3.2.1 Seismic Classification

- **Seismic Category I**
  - ✓ Based on RG 1.29 "Seismic Design Classification".
  - ✓ Designed to withstand the effects of the SSE and remain functional.
  - ✓ Required for all safety-related SSCs.

- **Seismic Category II**
  - ✓ SSCs not required to remain functional following an SSE but whose failure could degrade performance of safety-related SSCs to an unacceptable safety level.
  - ✓ Designed and constructed to maintain their structural integrity under seismic loading from the SSE.

3.2.1 Seismic Classification

- **Certain non-safety-related SSCs have special seismic design requirements, for example:**
  - ✓ Safety-related instrument lines used to actuate or monitor safety-related systems (RG 1.151)
  - ✓ Fire protection systems (RG 1.189)
  - ✓ Radioactive Waste Management Systems (RWMS) (RG 1.143)
  - ✓ Diverse Automatic Actuation System
- **Remaining SSCs assigned to Seismic Category NS.**

3.2.2 System Quality Group Classification

- **Quality Groups based on RG 1.26, NRC Quality Groups**
  - ✓ Quality Group A - ASME Code, Section III, Class 1, safety-related components that are part of the reactor coolant pressure boundary (RCPB)
  - ✓ Quality Group B -Pressure-retaining portions and supports not in Quality Group A for safety-related containment isolation, ECCS and residual heat removal functions
  - ✓ Quality Group C -Pressure-retaining portions and supports for other safety-related functions not included in Quality Groups A or B
  - ✓ Quality Group D -Non-safety-related water- and steam-containing components of systems that are not part of the RCPB or included in Quality Groups A, B or C, or RWMS, but are part of systems or portions of systems that contain or may contain radioactive material

3.2.2 System Quality Group Classification

- **Equipment Classes (1 through 10)**
  - ✓ Classes 1 through 3 - Safety-related systems and components
    - Correspond to RG 1.26 quality groups A, B, and C
    - Include other non-ASME class, safety-related components
  - ✓ Class 4 – Non-safety-related systems and components
    - Selected equipment addressed by NRC, Quality Group D
  - ✓ Class 5 – Non-safety-related components
    - Listed as “risk significant” in DCD Table 17.4-1
    - Designed to meet special seismic requirements such as seismic category II
    - Perform functions that address ATWS or station blackout
    - Are not within the purview of Equipment Classes 4, 6, 7, and 8

3.2.2 System Quality Group Classification

- **Equipment Classes (1 through 10)**
  - ✓ Class 6 – Non-safety-related components of the RWMS and a part of steam generator blowdown system
    - Meet RG 1.143
  - ✓ Class 7 – Non-safety-related components of the fire protection system
    - Meet RG 1.189
  - ✓ Class 8 – Non-safety-related components containing radioactive materials, and classified as Quality Group D that are not included in Equipment Classes 1-7
    - The applicable codes and standards are the same as Equipment Class 4
  - ✓ Classes 9 and 10 – Non-safety-related components and structures that do not fall into any one of Quality Groups A through D, or Equipment Classes 1 through 8

3.2.2 System Quality Group Classification

- **DCD Table 3.2-2 Lists Mechanical and Fluid SSCs, and Their Classification**
  - ✓ Seismic Category
  - ✓ Equipment Class
  - ✓ Quality Assurance Classification
    - 10 CFR 50 Appendix B ("Q") – safety-related SSCs, Equipment Classes 1 through 3
    - Augmented quality requirements (“A") – selected non-safety-related SSCs, Equipment Classes 4 through 7
      - Part III of the Quality Assurance Program (QAP) Description
  - ✓ Applicable Design Codes and Standards

3.2 Classification of Structures, Systems, and Components

➢ Major RAIs (Open Items)

RAI No.	Question	RAI Topic / NRC Concern	RAI Response / DCD Impact
1015-7054	03.09.03-31	RAI asked applicant to include in any future edition of "Revised Design Completion Plan for US-APWR Piping Systems and Components," the classifications from the response to RAI 580-4584, Question 3.2.2-16 regarding making design basis documents supporting quality group classification available for audit.	➢ RAI response is currently under preparation. ➢ Submitted "Updated Design Completion Plan for US-APWR Piping Systems and Components" on August 30, 2013 that addressed the staff's request.

Section 3.11:

Environmental Qualification of Mechanical and Electrical Equipment

3.11 Environmental Qualification of Mechanical and Electrical Equipment

Section No.	Description
3.11	Environmental Qualification of Mechanical and Electrical Equipment
3.11.1	Equipment Location and Environmental Conditions
3.11.2	Qualification Tests and Analyses
3.11.3	Qualification Test Results
3.11.4	Loss of Ventilation
3.11.5	Estimated Chemical and Radiation Environment
3.11.6	Qualification of Mechanical Equipment
3.11.7	Combined License Information
3.11.8	References
Appendix 3D	Equipment Qualification of Electrical and Mechanical Equipment

3.11 Environmental Qualification of Mechanical and Electrical Equipment

- **Purpose**
  - ✓ Environmental qualification (EQ) of mechanical and electrical equipment

- **Scope of Equipment in EQ Program**
  - ✓ Consistent with SRP 3.11 scope guidance
  - ✓ Consistent with RG 1.206 (C.I.3.11) scope guidance

3.11 Environmental Qualification of Mechanical and Electrical Equipment

- **EQ requirements:**
  - ✓ Qualified mechanical and electrical equipment
    - Designed to perform its safety function under all anticipated environmental conditions
    - Considered all anticipated environments for the required length of time
  - ✓ EQ of equipment located in a harsh environment
    - Demonstrated by testing and/or a combination of testing and analysis
  - ✓ QAP complies with 10CFR50 Appendix B
    - Ensures all requirements are accomplished

Appendix 3D Equipment Qualification of Electrical and Mechanical Equipment

- **DCD App.3D and US-APWR Equipment Qualification Program Technical Report (MUAP-08015) integrate:**
  - ✓ Functional Qualification of active mechanical equipment in DCD 3.9
  - ✓ Seismic Qualification in DCD 3.10
  - ✓ Environmental Qualification (EQ) in DCD 3.11
- **Table 3D-2 (US-APWR Equipment Qualification Equipment List) lists:**
  - ✓ Equipment required to be qualified under the Equipment Qualification Program
  - ✓ Active mechanical components required to be functionally qualified
  - ✓ Equipment with special seismic qualification requirements

Appendix 3D Equipment Qualification of Electrical and Mechanical Equipment

- **US-APWR Equipment Qualification Program Technical Report (MUAP-08015)**
  - ✓ Identifies the regulatory basis and industry standards
  - ✓ Provides EQ parameters
  - ✓ Describes the standard equipment qualification program, procedures and policies to be implemented during the design, procurement, construction, testing, turnover and operational phases of a project
  - ✓ Provides the requirements and guidance for development of a specific Project Equipment Qualification Program (PEQP)

3.11 Environmental Qualification of Mechanical and Electrical Equipment

➢ Major RAIs (Open Items) (1/3)

RAI No.	Question 3.11-XX	RAI Topic / NRC Concern	RAI Response / DCD Impact
589-4536	36,37,38	➢ Methodology and assumptions used to calculate TID after LOCA ➢ Update of Technical Report (MUAP-08015) to include containment airborne activity concentrations used to determine gamma and beta TID ➢ Mission dose for selected equipment that requires repair	➢ Revised Response to RAI 589-4536 Question 03.11-36 (including 37) and 38 was submitted in April, 2013. ➢ US-APWR Equipment Qualification Program Technical Report (MUAP-08015) will be updated to include methodology, assumptions and containment airborne activity concentrations used to calculate TID after LOCA. ➢ The changes for mission dose have been incorporated into DCD Revision 4 ➢ Status: Under NRC review
880-6142	42	➢ Equipment Qualification Data Package (EQDP) Template ➢ Information to be contained in the EQDP	➢ Revised Response to RAI 880-6142 Question 3.11-42 was submitted in February, 2013. ➢ US-APWR Equipment Qualification Program (MUAP-08015) will be revised to include the EQDP template as Attachment D of the report. ➢ Status: Under NRC review

3.11 Environmental Qualification of Mechanical and Electrical Equipment

➢ Major RAIs (Open Items) (2/3)

RAI No.	Question 3.11-XX	RAI Topic / NRC Concern	RAI Response / DCD Impact
805-5915, 880-6142	41,43	➢ The EQ program was consolidated with the US-APWR Equipment Qualification Program (MUAP-08015), but they are not addressing the same requirements. ➢ Use of the term "important to safety" to define the scope of the EQ program ➢ Completeness of Table 3D-2 as a list of equipment requiring environmental qualification	➢ Response to RAI 805-5915, Q 3.11-41 has revised Section 3.11 to include EQ, seismic qualification and functional qualification into US-APWR Equipment Qualification Program (MUAP-08015). ➢ Scope of EQ has been corrected to comply with the scope of SRP Section 3.11 in the response to RAI 1034-7055, Q 3.11-63, 64 and 65, submitted on June, 2013. ➢ Table 3D-2 has been corrected to add missing equipment or delete unnecessary equipment in the revised response to RAI 805-5915, submitted on February, 2013. ➢ The changes have been incorporated into DCD Revision 4. ➢ MUAP-08015 will be updated to reflect the changes. ➢ Status: Under NRC review

3.11 Environmental Qualification of Mechanical and Electrical Equipment

➢ Major RAIs (Open Items) (3/3)

RAI No.	Question 3.11-XX	RAI Topic / NRC Concern	RAI Response / DCD Impact
901-6257	55	➢ ITAAC to demonstrate EQ of electrical and active mechanical equipment.	➢ Existing ITAAC has been revised in the response to RAI 1031-7108, Q 3.11-62, submitted in August, 2013. ➢ The necessary DCD changes have been submitted to the NRC in attachments to the RAI response, and will be incorporated into DCD Rev 5. ➢ Status: Under NRC review
650-5093	39	➢ Commercial-grade dedication process to meet the 10CFR50.49 requirements.	➢ To clarify how CGD will be used in EQ, MHI will modify MUAP-08015, Section 3.1.1 to EQ of commercial grade item will be performed using a combination of qualification testing to the appropriate standards supplemented with an acceptable commercial grade dedication program and documentation as detailed NRC endorsed EPRI topical reports.

Section 3.3:
Wind, Tornado and Hurricane Loadings

3.3 Wind, Tornado and Hurricane Loadings

Subsection No.	Description
3.3.1	Wind Loadings
3.3.2	Tornado and Hurricane Loadings
3.3.3	Combined License Information

- **There are no Open Items in Section 3.3.**

3.3.1 Wind Loadings

- The design basis wind loadings are determined in accordance with American Society of Civil Engineers (ASCE)/Structural Engineering Institute (SEI), "Minimum Design Loads for Buildings and Other Structures", ASCE/SEI 7-05.

- The US-APWR severe wind speed is 155 mph (Annual probability of exceedance: 0.01)

3.3.2 Tornado and Hurricane Loadings

Design	Tornado	Hurricane
Maximum wind speed (mph)	230	160
Exceedance frequency (per year)	10^{-7}	10^{-7}
Atmosphere pressure drop (Psi)	1.2	Non
Missile	Automobile Sphere Pipe	Automobile Sphere Pipe
Regulatory Guide	RG 1.76	RG 1.221

3.3.2 Tornado and Hurricane Loadings

US-APWR

- **Tornado and hurricane velocity pressure loads are computed in accordance with procedures accepted by SRP 3.3.2.**
  - ✓ No tornado wind speed adjustment applies with respect to height.
- **Tornado atmospheric pressure loading is computed for maximum pressure drop.**
  - ✓ There is no atmospheric pressure effect for hurricane.
- **Load combinations are in accordance with procedures accepted by SRP 3.3.2 and are supplemented with the design criteria and procedures provided in BC-TOP-3-A.**

3.3 Wind, Tornado and Hurricane Loadings

➢ Major RAIs (Open Items)

- ✓ There are no Open Items in Section 3.3.

Section 3.5:
Missile Protection

3.5 Missile Protection

- **This section provides outline information on missile protection.**
- **The following are defined:**
  - ✓ Structures, Systems, and Components(SSCs) to be protected
  - ✓ Credible missile selection
  - ✓ Design for missile protection
- **There are two Open Items in this section.**

3.5 Missile Protection

Subsection No.	Description
3.5.1	Missile Selection and Description
3.5.2	Structures, Systems, and Components to be Protected from Externally Generated Missiles
3.5.3	Barrier Design Procedures
3.5.4	Combined License Information

3.5.1 Missile Selection and Description (1/3)

- **Six types of missiles are considered for US-APWR missile protection design:**
  - ✓ Internally Generated Missiles (Outside Containment)
  - ✓ Internally Generated Missiles (Inside Containment)
  - ✓ Turbine Missiles
  - ✓ Missiles Generated by Tornadoes and Hurricanes
  - ✓ Site Proximity Missiles (Except Aircraft)
  - ✓ Aircraft Hazards

- **Protection (probabilistic approach, physical separation, barrier design and design feature) for internally generated missiles is provided.**

3.5.1 Missile Selection and Description (2/3)

- **COL applicant is to identify site-specific SSCs to be protected from turbine missiles, and assess T/G orientation including for other units at multi-unit sites.**

3.5.1 Missile Selection and Description (3/3)

- **The design basis spectrum of Tornado and Hurricane missiles conforms to that defined in applicable regulatory guides:**
  - ✓ RG 1.76 for Tornado missiles
  - ✓ RG 1.221 for Hurricane missiles

- **The COL applicant will verify site interface parameters with respect to aircraft crashes and air transportation accidents.**

3.5.1.3 Turbine Missiles (1/4)

- **Two categories of turbine failures are evaluated:**
  - ✓ Design over-speed failure associated with brittle fracture
  - ✓ Destructive over-speed failures associated with ductile failure
- **Two supporting technical reports have been submitted:**
  - ✓ MUAP-07028 "Probability of Missile Generation from Low Pressure Turbines"
  - ✓ MUAP-07029 "Probabilistic Evaluation of Turbine Valve Test Frequency"

3.5.1.3 Turbine Missiles (2/4)

➢ **Turbine missile generation probabilities:**

✓ Brittle fracture within the range less than design over-speed

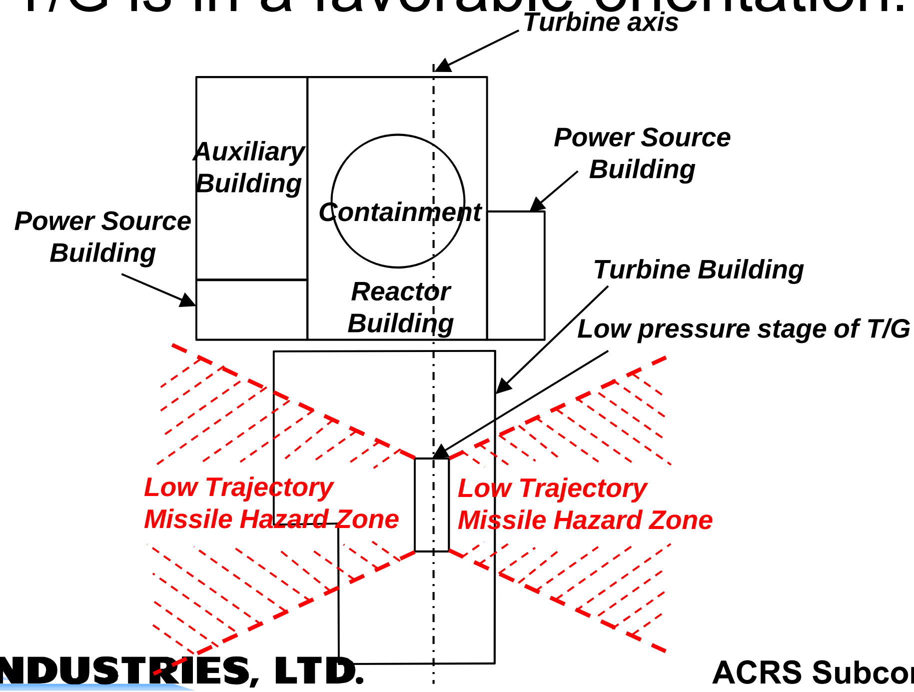
- Less than 10^{-5} per year unless ISI intervals exceeds 20 years (MUAP-07028)

✓ Destructive over-speed

- Less than 10^{-5} per year by quarterly turbine valve test frequency (MUAP-07029)

3.5.1.3 Turbine Missiles (3/4)

- **SRP 3.5.1.3 prescribes:**
  - ✓ P1 of favorably oriented turbine to be less than 10^{-4} to obtain P4 in acceptable risk rate of 10^{-7} per year.
- **US-APWR turbine is favorably oriented**
  - ✓ The R/B, PCCV, PS/B, and safety-related and non-safety related SSCs within these structures are located such that T/G is in a favorable orientation.



3.5.1.3 Turbine Missiles (4/4)

- **P1 for US-APWR turbine is maintained to be less than 10^{-5} by:**
  - ✓ Proper rotor design, material selection, PSI and ISI programs, and redundant protection and control system.
  - ✓ Details are discussed in Chapter 10, Section 10.2.2.3 and 10.2.3.

3.5.2 Structures, Systems, and Components to be Protected from Externally Generated Missiles

- **SSCs are protected from external missiles by the reinforced concrete external walls and roof of the safety related R/B and PS/B.**

3.5.3 Barrier Design Procedure

- Components, protective shields, and missile barriers are designed to prevent damage to safety-related components by absorbing and withstanding missile impact loads.
- The target SSCs, shields, and barriers are evaluated for both local effects and overall structural effects due to missile impacts.

3.5 Missile Protection

➢ Major RAIs (Open Items)

RAI No.	Question 03.05.03-X	RAI Topic / NRC Concern	RAI Response / DCD Impact
758-5680	10	The staff requested the applicant to provide an analysis assessing the local effects of an automobile missile on all seismic Category I structures.	➢ Submitted a response to the NRC on April 25, 2013. ➢ The response is under review by the NRC staff.

3.5 Missile Protection

➢ Major RAIs (Open Items)

RAI No.	Question	RAI Topic / NRC Concern	RAI Response / DCD Impact
782-5910, Rev. 3	14.03.07-58 (Open Item 3.5.1.3-4)	Plant System- Inspections, Tests, Analyses and Acceptance Criteria: NRC requests the followings regarding DCD Tier1 Section 2.7.1.1.1 (Design Description of T/G) and Table 2.7.1.1-1(T/G ITAAC)	
		➢ Key Design Features information regarding turbine orientation should be included in DCD Tier 1.	➢ MHI revised Tier 1 Section 2.7.1.1.1 to include the description of "turbine favorably oriented" with respect to the nuclear island within the standard design scope as suggested.
		➢ AC of ITAAC Table should state that the turbine missile probability analysis exists and concludes that the probability of missile is less than 10^{-5} per year.	➢ MHI revised to include AC in ITAAC Table stating that: • the turbine missile generation reconciliation report exists and concludes that all the as-built information such as LPT material properties are bounded by the assumptions of turbine missile analysis report. • turbine valve test frequency is consistent with that of valve test frequency evaluation report. ➢ MHI's response to a follow-up RAI (RAI1052-7205) is under review by the NRC staff.

Section 3.4:
Water Level (Flood) Design

3.4 Water Level (Flood) Design

Subsection No.	Description
3.4.1	Flood Protection
3.4.2	Analysis Procedures
3.4.3	Combined License Information

- **The US-APWR is designed to accommodate the effects of external or internal flooding**
  - ✓ Protection of plant nuclear safety functions during and after the internal or external flooding events is addressed in subsection 3.4.1.
  - ✓ Design conditions for Seismic Category I structures to withstand the hydrostatic/dynamic loads from the design-basis flood and/or groundwater conditions is addressed in subsection 3.4.2.

3.4.1 Flood Protection (1/4)

➢ External flood protection

✓ Water sources

- Probable maximum precipitation (PMP)
- Ground water
- Probable maximum flood (PMF) of streams and rivers
- Tsunami
- Surge, seiche flooding, and wave action
- Potential dam failures
- Outdoor water storage facility failures
- Potential cooling water canals and reservoir failures

✓ Design-basis flooding level (DBFL) for US-APWR standard design: 1 ft below the plant grade

✓ Site specific external flood conditions are addressed by COL applicants.

3.4.1 Flood Protection (2/4)

➢ External flood protection

✓ Key design features

- Thick reinforced concrete external walls and base mats
- Penetration with flood protection features
- Sloped roofs with drainage system

✓ Site specific design features (e.g. site grading and drainage, dikes, levees, retention basins) are addressed by COL applicants.

3.4.1 Flood Protection (3/4)

➢ Internal flood protection

- ✓ Water sources
  - Earthquakes (failure of non-seismic components)
  - Pipe breaks and cracks in accordance with Section 3.6.2
  - Fire fighting operations
  - Pump mechanical seal failures

3.4.1 Flood Protection (4/4)

- **Internal flood protection**
  - ✓ Key design features
    - Physical separation of safety-related SSCs by protective barriers, water-tight doors and penetration seals to preclude simultaneous loss of redundant systems
    - Enhanced piping design to minimize postulated flooding water source
    - Placement of safety-related SSCs above internal flood levels

- **The NRC staff conducted an audit of design documents in April, 2013.**

3.4.2 Analysis Procedures

- Discusses flooding effects to seismic category I structure design
- Section 3.8 provides the design and analysis procedures used to transform the static and dynamic effects of the DBFL and ground water levels applied to seismic category I structures.

3.4 Water Level (Flood) Design

➢ Major RAIs (Open Items)

RAI No.	Question 03.04.01-X	RAI Topic / NRC Concern	RAI Response / DCD Impact
841-6055	29	The RAI asked to provide the basis for assumptions made for evaluating flooding due to fire fighting operations, and to discuss how high and moderate energy line breaks and cracks are accounted for in the flooding analysis.	➢ Submitted the final response in July, 2013 after the technical audit in April, 2013, and the response is currently under evaluation. ➢ Provided the basis for assumptions for fire fighting operations and high or moderate line breaks in the flood evaluation. ➢ Internal flooding evaluation was updated to reflect a series of design changes. ➢ Updates and changes have been incorporated in DCD Rev.4.

3.4 Water Level (Flood) Design

➢ Major RAIs (Open Items)

RAI No.	Question 03.04.01-X	RAI Topic / NRC Concern	RAI Response / DCD Impact
841-6055	30	The RAI asked to update Figure 3K-5 to show flood barriers at the main control room vestibule.	➢ Submitted a response in October 2012 where MHI stated that the figure would be updated in accordance with the closure plan for seismic and structural analyses. ➢ The updated figure has been incorporated in DCD Rev.4.

3.4 Water Level (Flood) Design

➢ Major RAIs (Open Items)

RAI No.	Question 03.04.02-X	RAI Topic / NRC Concern	RAI Response / DCD Impact
546-4345 489-3516	5 and 6	Requests to further clarify the use of 0.7 as the coefficient of friction at the soil-concrete interface.	➢ MUAP-12002 Revision 1 "Sliding Evaluation and Results", which utilized new coefficient of friction 0.5, was submitted in January 2013. ➢ Revised responses to Question 03.04.02-5 and 6 were submitted in March 2013. ➢ These documents are currently under review by the NRC staff.

Section 3.6:
Protection Against Dynamic Effects Associated with Postulated Rupture of Piping

3.6 Protection Against Dynamic Effects Associated with Postulated Rupture of Piping

- **This section provides outline information on the pipe rupture protection.**

- **The following are defined for the US-APWR:**
  - ✓ Design measures and design bases
  - ✓ Postulated pipe rupture location and jet impingement load
  - ✓ Leak-Before-Break (LBB) evaluation methodology

- **This section has no open items.**

3.6 Protection Against Dynamic Effects Associated with Postulated Rupture of Piping

Subsection No.	Description
3.6.1	Plant Design for Protection against Postulated Piping Failure in Fluid Systems Inside and Outside Containment
3.6.2	Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping
3.6.3	LBB Evaluation Procedures
3.6.4	Combined License Information
Appendix-3B	Bounding Analysis Curve Development for Leak Before Break Evaluation of High-Energy Piping for US-APWR
Appendix-3E	High-Energy and Moderate Energy Piping in the Prestressed Concrete Containment Vessel and Reactor Building

3.6.1 Plant Design for Protection Against Postulated Piping Failure in Fluid Systems Inside and Outside Containment

- **In accordance with SRP 3.6.1, the plant is designed to provide protection against piping failure to ensure that such failures would not compromise the functional capability of safety-related systems.**

- **In order to maintain the safety of the plant when a pipe break is postulated, required plant conditions, design measure and design basis are provided.**

3.6.2 Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

- **Break or crack location is postulated in accordance with SRP BTP 3-4.(eg ; high stress location, location that has great effect on essential equipment, etc.).**
- **Jet thrust reaction force and jet impingement load are defined in accordance with SRP 3.6.2.**
- **Design criteria for pipe whip restraint and jet impingement barriers are provided.**

3.6.3 LBB Evaluation Procedures

- This subsection describes the design basis to eliminate the dynamic effects of pipe rupture for high-energy piping systems of RCL piping, RCL branch piping, and main steam piping.

- LBB evaluation is performed in accordance with SRP 3.6.3.

- LBB criteria are applied to RCL piping, RCL branch piping with normal diameter of 6 inches or larger, and Main Steam piping in PCCV.

3.6 Protection Against Dynamic Effects Associated with Postulated Rupture of Piping

- **Major RAIs (Open Items)**
  - ✓ There are no Open Items in Section 3.6.

Subsection 3.9.1: Special Topics for Mechanical Components

3.9.1 Special Topics for Mechanical Components

- This subsection provides information on methods of analysis for ASME Code Section III, Division 1, Class 1, 2, and 3 components and supports including core support structures.

- There are no Open Items in this section.

3.9.1 Special Topics for Mechanical Components

Subsection No.	Description
3.9.1.1	Design Transients
3.9.1.2	Computer Program Used in Analyses

3.9.1.1 Design Transients

- **Components are evaluated using the design transients in accordance with the requirements for Class 1 in ASME Code, Section III.**
- **The design transients give fluid system pressure, temperature, flow transients, and frequency to perform the ASME Code fatigue analysis and stress analysis.**
  - ✓ These do not cover the seismic loading and other mechanical loading on each component.
- **The design transient item of each Service Level (A, B,C and D) and test condition is addressed.**
- **The 60 year design life is considered when determining the number of occurrences of each transient.**

3.9.1.2 Computer Program Used in Analyses

- A number of computer programs which are commercial codes and house codes, are used for static, dynamic, and hydraulic transient analysis for component design.

- All computer programs are verified and validated in accordance with ASME NQA-1, Subpart 2.7.

- NRC audit for computer programs was completed in August 2011 and closed.

➢ Major RAIs (Open Items)

✓ There are no Open Items in Subsection 3.9.1.

Subsection 3.9.2:
Dynamic Testing and Analysis of Systems, Components, and Equipment

3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment

Subsection No.	Description
3.9.2.1	Piping Vibration, Thermal Expansion, and Dynamic Effects
3.9.2.2	Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment
3.9.2.3	Dynamic Response Analysis of Reactor Internals under Operational Flow Transients and Steady-State Conditions
3.9.2.4	Preoperational Flow-Induced Vibration Testing of Reactor Internals
3.9.2.5	Dynamic System Analysis of the Reactor Internals under Faulted Conditions
3.9.2.6	Correlations of Reactor Internals Vibration Tests with the Analytical Results

3.9.2 Dynamic Testing and Analysis of Systems Components and Equipment

US-APWR

- **Dynamic Analysis Methodology (Subsections 3.9.2.1, 3.9.2.2 and 3.9.2.5)**
  - ✓ A Finite Element (FE) model of the Reactor Coolant Loop (RCL) is used for analysis of RCL system response.
  - ✓ Additional FE sub-system models of SG and Reactor Pressure Vessel are applied for analysis of internal structures.
  - ✓ Dynamic hydraulic loads during LOCA event are evaluated by blow-down analysis with ‘MULTIFLEX’ code.

3.9.2 Dynamic Testing and Analysis of Systems Components and Equipment

- **Flow-Induced Vibration (FIV) Assessment for Reactor Internals (3.9.2.3, 3.9.2.4, 3.9.2.6 and TeR MUAP-07027)**
  - ✓ FIV assessment program is established in accordance with RG1.20.
  - ✓ Conclusion of Prediction analysis: Reactor Internals have sufficient margins for adverse flow effects (Fluid Elastic Instability, Vortex-shedding lock-in, Turbulence / RCP pulsation-induced vibration).
  - ✓ Test program for the first operating plant: Vibration measurement and high-cycle fatigue evaluation will be conducted in Hot Functional Test (HFT).

3.9.2 Dynamic Testing and Analysis of Systems, Components and Equipment

➢ Major RAIs (Open Items)

RAI No.	Question 03.09.02-X	RAI Topic / NRC Concern	RAI Response / DCD Impact
1013 - 7011	103	(a) Provide a comparison of technical data of both SGs (SONGS and US-APWR) (b) Explain any design differences between the SG of the US-APWR and that of Fort Calhoun. (c) Explain how the contact force of the AVBs will be checked to ensure it is sufficiently high to prevent in-plane tube instability. (d) Explain why the wear at the tube support plates is considered to be caused by turbulence excitation and not by in-plane tube instability.	➢ Response is under preparation

3.9.2 Dynamic Testing and Analysis of Systems, Components and Equipment

➢ Major RAIs (Open Items)

RAI No.	Question 03.09.02-X	RAI Topic / NRC Concern	RAI Response / DCD Impact
1013 - 7011	104	Provide the preliminary design of the SG tube bundle and the design criteria for the SG tubes and retainer bars against flow-induced excitations, including random turbulence, fluid elastic instability, and vortex shedding.	➢ Response is under preparation.

Subsection 3.9.3:
ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structure

Subsection No.	Description
3.9.3.1	Load Combinations, System Operating Transients, and Stress Limits
3.9.3.2	Design and Installation of Pressure-Relief Devices
3.9.3.3	Pump and Valve Operability Assurance
3.9.3.4	Component Supports
Appendix 3C	Reactor Coolant Loop Analysis Methods

- **There are two Open Items in this subsection**

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structure

- **This subsection provides regulatory compliance with the following.**
  - ✓ 10 CFR 50, Appendix A and 10 CFR 50.55a for the design of safety-related structure and components.
  - ✓ 10 CFR 50, Appendix S for seismic category I structure and components.
  - ✓ GDC 2 for seismic design, effect of the earthquake without loss of capability to perform safety function.
  - ✓ GDC 4 for normal and accident environmental conditions.
  - ✓ GDC 14 and GDC 15 for RCPB design of safety-related components in various operating conditions in compliance with ASME Code, Section III requirements.

3.9.3.1 Load Combinations, System Operating Transients, and Stress Limits

➢ Component Design

- ✓ Component design is performed in accordance with ASME B&PV Code.
- ✓ Load combinations for all ASME Service Levels, such as dynamic loads (seismic, accident) and static loads (dead weight, thermal load) are used.
- ✓ Design criteria and stress limits are based on ASME Code, Section III.
- ✓ Reactor Coolant Loop design is presented in Appendix 3C and Technical Report MUAP-09002, "Summary of Seismic and Accident Load Conditions for Primary Components and Piping".
- ✓ Design specifications for Risk Significant ASME Class 1,2,3 PSCs including supports are provided in accordance with ASME Section III.

3.9.3.2 Design and Installation of Pressure-Relief Devices

➢ Design and Installation of Pressure-Relief Devices

- ✓ Pressure relief valves are designed in compliance with the requirements of ASME Code, Section III.
- ✓ Pressurizer safety valves provide over-pressure protection for the RCS.
- ✓ Safety valves and power operated relief valves are provided on the steam lines.
- ✓ Relief valves are provided on the RHRS. These also perform low temperature over-pressure protection for the RCS.

3.9.3.3 Pump and Valve Operability Assurance

➢ Pump Operability

- ✓ The operability of active pumps is established for safety-related functions in operating conditions.
- ✓ Active pump design criteria and stress limits for pressure-retaining and support functions are based on ASME Code, Section III.

➢ Valve Operability

- ✓ The operability of active valve is established for safety-related functions in operating conditions.
- ✓ Active valve design criteria and stress limits for pressure-retaining and support functions are based on ASME Code, Section III.

3.9.3.4 Component Supports

➢ Component Support Design

- ✓ Component support design is confirmed to be in accordance with ASME B&PV Code.
- ✓ Load combinations for all ASME Service Levels
- ✓ Design criteria and stress limits are based on ASME Section III.
- ✓ Support classification into two classifications.
  1. Manufactured standard supports
     - Spring Hanger
     - Snubber
     - Strut
  2. Supplementary steel supports
     - Frame type pipe supports
     - Base plates and anchor bolts

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structure

➢ Major RAIs (Open Items)

RAI No.	Question	RAI Topic / NRC Concern	RAI Response / DCD Impact
209-1803	03.09.03-21	Requests details of the modeling and analysis methods of supports used for major components, including sketches of the support design, loads and load combinations, applicable stress limit criteria, and the fatigue evaluation criteria.	➢ Submitted a response on April 30, 2009 ➢ The requested details were provided in MUAP-08005, which was superseded by MUAP-10006 and 09002 submitted on November 30, 2012 and July 31, 2013, respectively. ➢ Fatigue evaluation will be performed in accordance with ASME Code, Section III Subsection NF. ➢ This RAI is an Open Item pending the staff's review of the two reports above.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structure

➢ Major RAIs (Open Items)

RAI No.	Question	RAI Topic / NRC Concern	RAI Response / DCD Impact
1015 - 7054	03.09.03-31	Requests MHI to inform the staff when the PSC design specifications within the audit scope become available and the staff may commence its audit.	➢ Submitted "Revised Design Completion Plan for US-APWR Piping Systems and Components" on December 7, 2012 and its revision on August 30, 2013, both of which requested the audit be planned for December 2013. ➢ MHI will submit a response to confirm completion of the design specifications.

Subsection 3.9.4:
Control Rod Drive Systems

3.9.4 Control Rod Drive Systems

Subsection No.	Description
3.9.4.1	Descriptive Information of CRDS
3.9.4.2	Applicable CRDS Design Specifications
3.9.4.3	Design Loads, Stress Limits, and Allowable Deformations
3.9.4.4	CRDS Operability Assurance Program

3.9.4 Control Rod Drive Systems

- This subsection provides information on design, functional requirements, and operability assurance program for the control rod drive mechanism (CRDM) portion of the CRDS.

- The CRDM for the US-APWR is of the magnetically operated jacking type.
  - ✓ Based on the L-106A type CRDM which has been used in many operating plants in the United States and Japan.

- The CRDM pressure housing is designed in accordance with ASME Code, Section III, Subsection NB.

3.9.4 Control Rod Drive Systems

➢ Major RAIs (Open Items)

RAI No.	Question 03.09.4-X	RAI Topic / NRC Concern	RAI Response / DCD Impact
107-1293	1 (Sub-questions 1293-01, 1293-06, and 1293-07)	Requests to show the basis of the allowable rod travel housing deflection and demonstrate that the estimated CRDM deflection for the US-APWR does not exceed the allowable limit.	➢ In the amended response , MHI shows that estimated deflection of the CRDM pressure housing at Level D condition is within the allowable limit. ➢ The responses to these sub-questions will be revised to reflect updated seismic information. ➢ No impact on DCD is expected.
848-6093	14	Requests to provide the justification for the increase in the maximum CRDM deflection for the level D condition in the amended response to RAI No.107-1293.	➢ MHI explained that the CRDM dynamic response analysis results reflected the seismic conditions of US-APWR DCD Rev.3. ➢ The response to this question will be revised reflecting updated seismic information. ➢ No impact on DCD is expected.

Subsection 3.9.5:
Reactor Pressure Vessel Internals

3.9.5 Reactor Pressure Vessel Internals

Subsection No.	Description
3.9.5.1	Design Arrangements
3.9.5.2	Loading Conditions
3.9.5.3	Design Bases

3.9.5 Reactor Pressure Vessel Internals

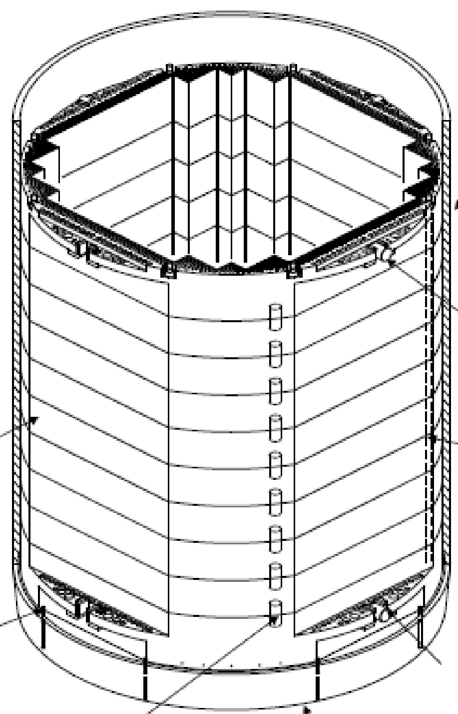
US-APWR

- This subsection provides outline information on the structure and the design bases of Upper and Lower Reactor Pressure Vessel Internals (RIs).

- The codes and standards applicable to Reactor Internals are:
  - ✓ ASME Code, Section III, Subsection NG
  - ✓ ASME Code, Section XI

3.9.5 Reactor Pressure Vessel Internals

- **US-APWR adopts the use of a Neutron Reflector, replacing conventional baffle structures.**
  - ✓ Improvement in neutron reflectivity
  - ✓ Significant reduction in the number of threaded fasteners



3.9.5 Reactor Pressure Vessel Internals

US-APWR

➢ Major RAIs (Open Items)

✓ There are no Open Items in Section 3.9.5

Subsection 3.9.6:

Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints

3.9.6 Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints

Subsection No.	Description
3.9.6.1	Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints
3.9.6.2	IST Program for Pumps
3.9.6.3	IST Program for Valves
3.9.6.4	IST Program for Dynamic Restraints
3.9.6.5	Relief Request and Authorization to ASME OM Code

- **ASME Code, Section III, Class 1, 2 and 3 safety-related pumps, valves and dynamic restraints that are required to perform a safety function, are subjected to IST to assess and verify operational readiness as set forth in 10 CFR 50.55a(f) and ASME OM Code.**

- **IST Program implements ASME OM Code 2004 Edition through 2006 addenda.**

- **The COL Applicant is responsible for administrative control of the IST program as well as control of the ASME OM Code edition and addenda to be used for their IST program.**

Major RAIs (Open Items)

RAI No.	Question	RAI Topic / NRC Concern	RAI Response / DCD Impact
288-2274	03.09.06-1	This RAI is an Open Item to track an audit of the US-APWR design and procurement specifications to evaluate implementation of the functional design, qualification, and IST programs.	➢ Preparation of sample design and procurement specifications is currently in progress.

Section 3.10: Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

3.10 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

US-APWR

Subsection No.	Description
3.10.1	Seismic Qualification Criteria
3.10.2	Methods and Procedures for Qualifying Mechanical and Electrical Equipment and Instrumentation
3.10.3	Methods and Procedures of Analysis or Testing of Supports of Mechanical and Electrical Equipment and Instrumentation
3.10.4	Test and Analyses Results and Experience Database
3.10.5	Combined License Information

3.10.1 Seismic Qualification Criteria (1/2)

US-APWR

➢ Qualification Standards

- ✓ IEEE Std 344-2004, as modified by RG 1.100, for safety-related mechanical and electrical equipment and their supports
- ✓ ASME Code, Section III for structural integrity of safety-related pressure boundary components
- ✓ ASME QME-1-2007 for qualification of active mechanical equipment

3.10.1 Seismic Qualification Criteria (2/2)

- **Performance Requirements for:**
  - ✓ Seismic Category I instrumentation and electrical equipment
    - Provided in corresponding Equipment Qualification Summary Data Sheet (EQSDS)
  - ✓ Seismic Category I active mechanical components
    - Defined in corresponding equipment specifications along with system functional requirements
- **Performance Criteria**
  - ✓ To perform their designated safety-related function(s) under the postulated SSE in combination with other concurrent loadings
  - ✓ Deformation of supports and structures is acceptable, provided their and/or other equipment's safety-related functional performance are not compromised

3.10.2 Methods and Procedures for Qualifying Mechanical and Electrical Equipment and Instrumentation

- **Testing**
  - ✓ In accordance with RG 1.100 Rev.3 (IEEE Std 344-2004, ASME QME-1-2007).
  - ✓ EQ Program provided in DCD Section 3.11.
- **Analysis**
  - ✓ In accordance with RG 1.100 Rev.3 (IEEE Std 344-2004, ASME QME-1-2007).
  - ✓ Analysis without testing is acceptable only if structural integrity alone can assure the design-intended design function.
- **Combination of Testing and Analysis**
  - ✓ In accordance with RG 1.100 Rev.3 (IEEE Std 344-2004, ASME QME-1-2007).
  - ✓ Utilized when the equipment cannot be practically qualified by analysis or testing alone

3.10.3 Methods and Procedures of Analysis or Testing of Supports of Mechanical and Electrical Equipment and Instrumentation

- **Tests or analyses to assure structural capability, including anchorage (See 3.10.2)**
  - ✓ Electrical equipment and instrumentation supports (including instrument racks, control consoles, cabinets, and panels) are tested with the equipment installed or equivalent dummy.
  - ✓ For mechanical equipment supports (including pumps, valves, valve operators and fans), in accordance with ASME Code, Section III
  - ✓ For instrumentation line supports, using the criteria from ASME Code, Section III, Subsection NF for Equipment Class 1 and 2 supports

3.10.4 Test and Analyses Results and Experience Database

- **Equipment qualification file includes:**
  - ✓ Qualification method used for equipment
  - ✓ Tests and analyses results
  - ✓ List of systems, equipment
  - ✓ Equipment support structures
  - ✓ EQSDSs, which summarize the component's qualification
  - ✓ Seismic Input Requirements

- **Experience Based Qualification**
  - ✓ Not used for any equipment

3.10 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

➢ Major RAIs (Open Items)

RAI No.	Question	RAI Topic / NRC Concern	RAI Response / DCD Impact
486-3861	03.10-17	Requests to (1) provide a list of components in the GTG System to be seismically qualified with the method of seismic qualification specified for each component and estimated qualification schedules so that the staff will have opportunities to witness the testing; and (2) describe, in DCD Tier 2, Section 3.10, the seismic qualification criteria and procedures including referenced report number for related electrical and mechanical components of the GTG System.	➢ Submitted a revised response in June 2013. ➢ This RAI is an Open Item pending the submittal and the staff review of MUAP-10023 Revisions. • Revision 6 of MUAP-10023 was submitted in September 2013. • Next revision is planned for December 2013.

Section 3.12:

Piping Design Review

3.12 Piping Design Review

Subsection No.	Description
3.12.1	Introduction
3.12.2	Codes and Standards
3.12.3	Piping Analysis Methods
3.12.4	Piping Modeling Technique
3.12.5	Piping Stress Analysis Criteria
3.12.6	Piping Support Design Criteria

3.12.1 Introduction

- This section covers the design of the piping systems including piping supports which comprise seismic category I and non-seismic category I (seismic category II and non-seismic) piping systems.

3.12.2 Codes and Standards

- Codes and standards are consistent with 10 CFR 50, Appendix A, GDCs 1, 2, 4, 14, and 15, and 10 CFR 50, Appendix S.
- Piping analysis uses the 1992 Edition with 1992 Addenda of the ASME Code Section III, Division 1, Subsections NB, NC, and ND in accordance with the requirements of 10 CFR 50.55a.
- Material properties comply with ASME Code Section II, 2001 Edition including 2003 Addenda.

3.12.3 Piping Analysis Methods

- **Seismic analysis for all seismic category I and II piping systems use methods in accordance with SRP 3.7.3.**

- **These methods include the response spectrum method or where applicable, the equivalent static load method.**

- **For modeling supports in the piping analysis, the de-coupled support model (modeled as stiffness) or an integrated support model (the actual structural model) of the support is used.**

3.12.4 Piping Modeling Technique

- **Dynamic analysis (3D space frame) consists of:**
  - ✓ A sequence of nodes connected by straight pipe elements
  - ✓ Curved pipe elements with stiffness properties representing the piping
  - ✓ Other inline components.

- **Computer programs are verified and validated using NUREG/CR-1677, Volumes 1 and 2.**

- **NRC audit for computer program and design methodology documented in DCD 3.12 was completed in August 2011.**

3.12.5 Piping Stress Analysis Criteria (1/2)

- Allowable stresses of piping system for various Loads and Load Combinations are defined based on ASME Code Section III, NB/NC/ND.

- Combination of modal response (SRSS, 10% grouping method) and high frequency modes are considered in accordance with RG 1.92.

- Class 1 piping is evaluated for the effects of fatigue caused by thermal transients and other cyclic events including earthquakes and thermal stratification occurring in the Surge Line.

3.12.5 Piping Stress Analysis Criteria (2/2)

- Thermal stratification and oscillation in the closed branch piping connected to RCS would be prevented in accordance with US-APWR design approach described in DCD Subsection 3.12.5.9 (Rev.4)

- The environmental impact on fatigue of Class 1 piping follow the requirements delineated in RG 1.207.

3.12.6 Piping Support Design Criteria

- Seismic category I pipe supports are designed in accordance with subsection NF of the ASME Code Section III, 2001 Edition including 2003 Addenda.

- The load combinations for the piping support design are defined based on Level A, B, C, and D service conditions.

3.12 Piping Design Review

➢ Major RAIs (Open Items)

RAI No.	Question 3.12-XX	RAI Topic / NRC Concern	RAI Response / DCD Impact
804-5938	26	Requests MHI to clarify the analysis model and input for reactor coolant loop piping (RCL) dynamic analysis.	➢ Submitted a response with the requested clarification in September 2012. ➢ The seismic and accident design analysis method for RCL system components is presented in MUAP-09002. The next revision of the report will be submitted in November 2013. ➢ There are DCD changes (3.7.1, 3.7.2, and 3.7.3) stated in the response that have been incorporated into DCD Rev4 and that need to be reviewed by the NRC staff. ➢ Therefore, this RAI is an Open Item pending the staff's review of MUAP-09002 and DCD Rev 4. However, the design of the RCL piping is covered by Subsection 3.9.3, and therefore, this open item is not technically concerned with Section 3.12.

Section 3.13:
Threaded Fasteners
(ASME Code Class 1, 2, and 3)

3.13 Threaded Fasteners (ASME Code Class 1, 2, and 3)

Subsection No.	Description
3.13.1	Design Considerations
3.13.2	Inservice Inspection Requirement
3.13.3	Combined License Information

3.13 Threaded Fasteners (ASME Code Class 1, 2, and 3)

- **Design Requirement – Same as that for Existing Plants**
  - ✓ Material selection: ASME B&PV Code
    - Selection Criteria: in accordance with NCA-1220 and NB/NC/ND-2128
    - Fracture Toughness: NB/NC/ND-2300, 10 CFR 50 App-G
  - ✓ Lubricant:
    - Fel-Pro, Neolube #126 and Nuclear Grade Neverseez are mainly used
    - MoS2 and Copper-based anti-seize compounds are not to be used
  - ✓ Reactor Vessel Closure Stud Bolt requirements
- **ISI Requirement**
  - ✓ ISI procedure is based on ASME Code, Section XI.

3.13 Threaded Fasteners (ASME Code Class 1, 2, and 3)

- **Major RAIs (Open Items)**
  - ✓ There are no Open Items in this section.

Acronyms (1/3)

AC	Acceptance Criteria
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
B&PV	Boiler and Pressure Vessel
BTP	Branch Technical Position
CFR	Code of Federal Regulation
CGD	Commercial Grade Dedication
CRDM	Control Rod Drive Mechanism
CRDS	Control Rod Drive System
DBFL	Design-basis Flooding
DCD	Design Control Document
ECCS	Emergency Core Cooling System
EQ	Environmental Qualification
EQDP	Equipment Qualification Data Package
EQSDS	Equipment Qualification Summary Data Sheet
GDC	General Design Criteria
HFT	Hot Functional Test
ISI	Inservice Inspection
IST	Inservice Testing
ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria

Acronyms (2/3)

LBB	Leak-Before-Break
LOCA	Loss of Coolant Accident
NRC	Nuclear Regulatory Commission
PCCV	Prestressed Concrete Containment Vessel
PEQP	Project Equipment Qualification Program
PMF	Probable Maximum Flood
PMP	Probable Maximum Precipitation
PS/B	Power Source Building
PSC	Piping Systems and Components
PSI	Preservice Inspection
QAP	Quality Assurance Program
QG	Quality Group
R/B	Reactor Building
RCL	Reactor Coolant Loop
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RG	Regulatory Guide
RHRS	Residual Heat Removal System
RI	Reactor Internal
RWMS	Radioactive Waste Management System

Acronyms (3/3)

SEI	Structural Engineering Institute
SG	Steam Generator
SRP	Standard Review Plan
SRSS	Square Root Sum of the Squares
SSC	Structures, Systems, and Components
SSE	Safe Shutdown Earthquake
T/G	Turbine Generator
TID	Total Integrated Dose

Backup Slides

LCF Analysis

➢ Calculation Procedure

- ✓ Crack growth rate is governed by Paris Eq.

  $$(da/dN) = C_0(\Delta K)^n$$

da/dN:	Crack growth rate per cycle
ΔK:	Stress Intensity Range ($\propto \Delta\sigma\sqrt{\pi \cdot a}$)
C_0, n:	Parameter (depends on material)

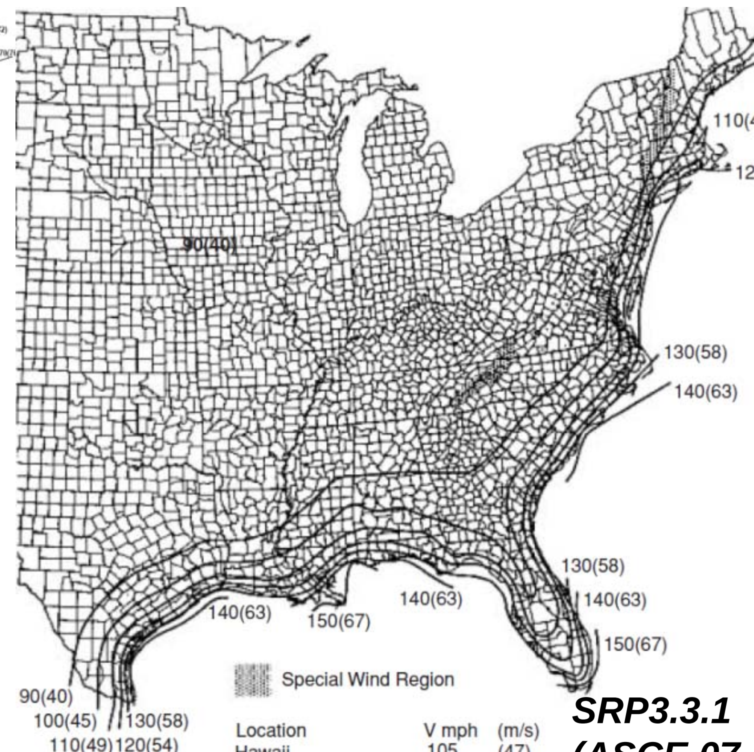
- ✓ Relation between the number of stress cycle due to startup/stop (N_f) and critical crack size (a_{cr})

  $$N_f = (2/((n-2)C_0 M^{n/2} \Delta\sigma^n)) x (a_i^{-(n-2)/2} - a_{cr}^{-(n-2)/2})$$

a_i:	Initial crack size
a_{cr}:	critical crack size (= $(Q/\pi)\cdot(K_{IC}/\sigma)^2$)
Δσ:	Range of stress cycles in operation
K_{IC}:	Fracture toughness
M:	π/Q
Q:	Flaw shape parameter

Severe Wind v.s. Extreme Wind

RG1.221

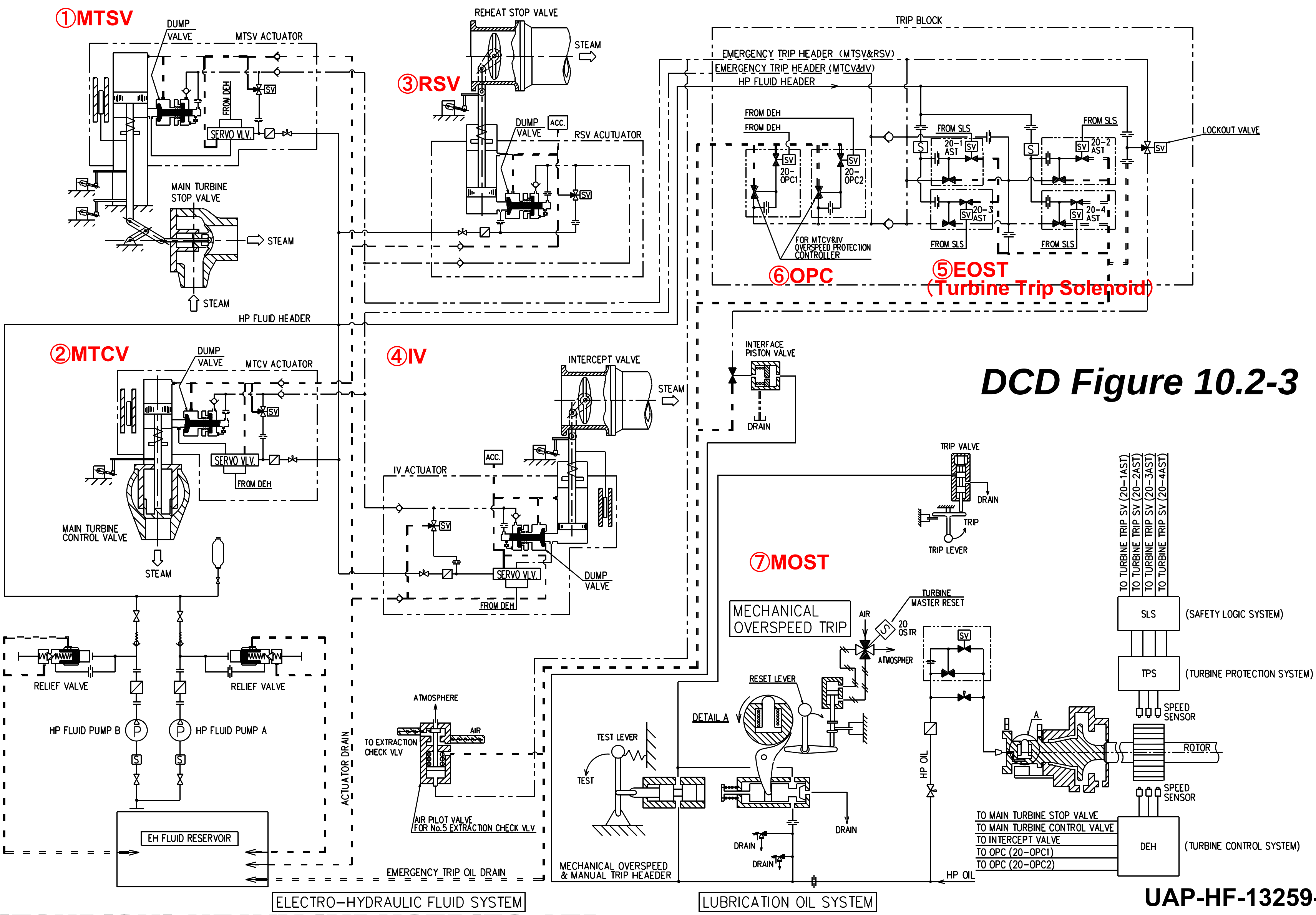


SRP3.3.1
(ASCE 07-05)

160mph
1E-07 Annual Exceedance

155mph
0.01 Annual Exceedance

Control and Protection System Diagram

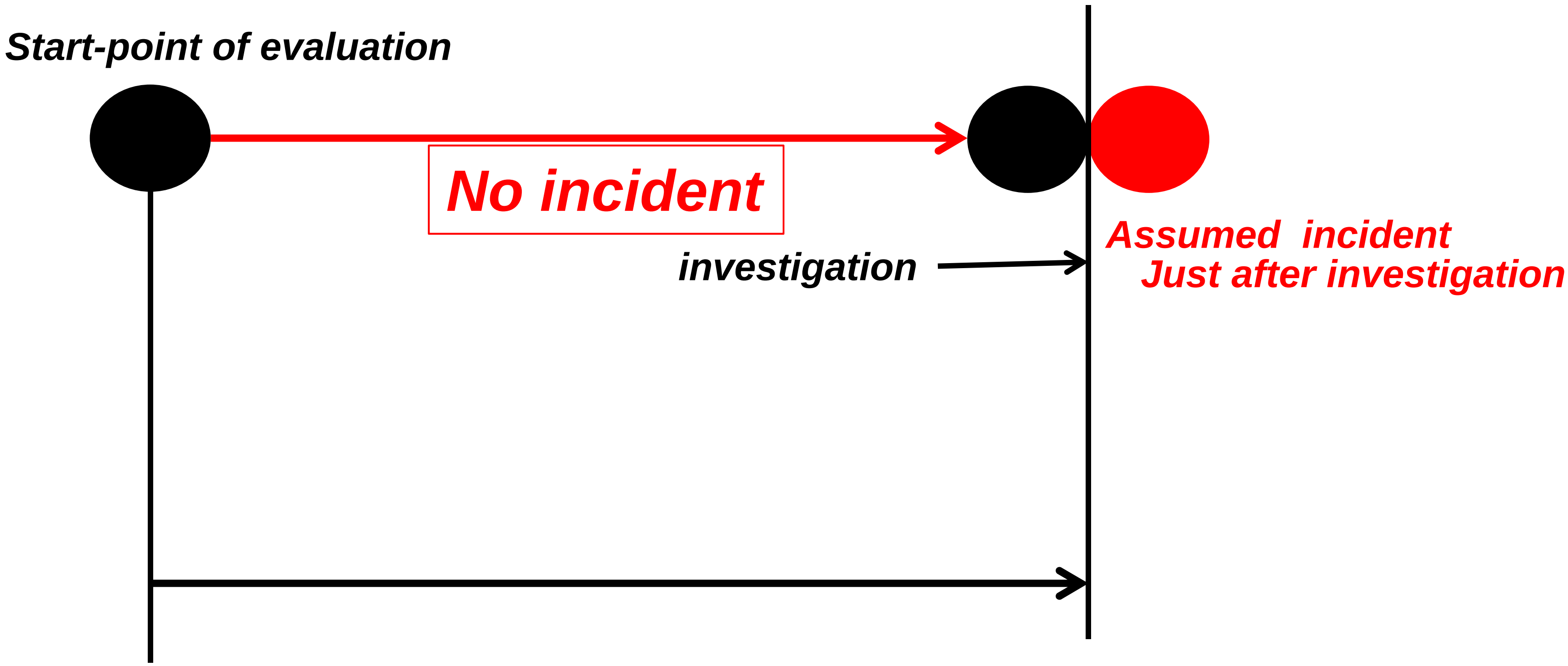


DCD Figure 10.2-3

Data and Conservative Assumptions (MUAP07029 Table 5.1-1)

- **Data taken from All Japanese PWR units**
  - ✓ Total accumulated operating hours of the units
- **Such incidents as below are counted as valve or system failure**
  - ✓ Servo vlv. malfunction due to degraded oil
  - ✓ Servo vlv. or SV malfunction due to inadequate maintenance
  - ✓ Single failure of the component such as LVDT
- **No actual unexpected turbine overspeed by incidents observed during investigation period**
- **All the incidents including any single failure of the components within the entire system are counted as MTCV or protection system failure for conservative purpose**

Estimation of the failure rate probability



Using this no incident period and assumed incident just after incident, estimate upper limit of the probability

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Presentation to the ACRS Subcommittee

MHI US-APWR Design Certification Application Review

Safety Evaluation Report with Open Items

Chapter 3: Design of Structures, Components, Equipment and Systems

November 20-21, 2013

Staff Review Team

- **Mechanical Engineering Branch**
  - **Yiu Law – Section 3.2.1**
  - **Sardar Ahmed – Section 3.2.2**
  - **Renee Li – Section 3.6.2**
  - **John Wu – Section 3.9.1**
  - **Yuken Wong – Section 3.9.2**
  - **Tuan Le – Section 3.9.3**
  - **Jason Huang – Sections 3.9.4, 3.9.5**
  - **Thomas Scarbrough – Section 3.9.6**
  - **P.Y. Chen – Section 3.10**
  - **James Strnisha – Section 3.11, 3.13**
  - **Robert Hsu – Section 3.12**
- **Structural Engineering Branch 1**
  - **Jerry Chuang – Sections 3.3.1, 3.3.2, 3.4.2, 3.5.3**

Staff Review Team

- **Balance of Plant and Fire Protection Branch**
  - **Angelo Stubbs – Section 3.4.1**
  - **Ryan Nolan – Sections 3.5.1.1, 3.5.1.2, 3.5.1.4, 3.5.2**
  - **Raul Hernandez – Section 3.6.1**
- **Materials and Chemical Engineering Branch**
  - **John Honcharik – Section 3.5.1.3**
  - **Eric Reichelt – Sections 3.6.3**
- **Radiation Protection and Accident Consequences Branch**
  - **Rao Tammara – Section 3.5.1.5, 3.5.1.6**
  - **Ronald LaVera – Section 3.11**
- **Electrical Engineering Branch**
  - **Peter Kang – Section 3.11**

- **Project Managers**
  - **Lead PM: Perry Buckberg**
  - **Chapter PM: Dennis Galvin**

Chapter 3 Review Status

SRP Section/Application Section		Number of OI
3.2	Classification of Structures, Systems, and Components	2
3.3	Wind and Tornado Loadings	0
3.4	Water Level (Flood) Design	3
3.5	Missile Protection	2
3.6	Protection Against Dynamic Effects Associated with Postulated Rupture of Piping	0
3.9	Mechanical Systems and Components	7
3.10	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	1
3.11	Environmental Qualification of Mechanical and Electrical Equipment	8
3.12	ASME Code Class 1, 2, and 3 Piping Systems, Piping Components, and their Associated Supports	1
3.13	Threaded Fasteners (ASME Code Class 1, 2, and 3)	0
Totals		24

- The SE excludes sections 3.7 and 3.8, which are still under review.

Description of Open Items

- RAI 1015-7054, Question 03.09.03-31: This RAI tracks the need for MHI to make available for audit design specifications and other design documents for ASME Class 1, 2, and 3 Components. The open item applies to Sections 3.2.1, 3.2.2, and 3.9.3. The audit will also address topics related to Sections 3.9.4, 3.9.5, 3.9.6, and 3.11.
- Status: Open Item to be closed upon the staff completing and documenting the audit, currently planned for February 2014.

- RAI 841-6055, Question 03.04.01-29: This RAI tracks the need for the staff to complete a flooding analysis audit concerning modeling assumptions and pipe break selections.
- Status: The staff has completed the audit and is reviewing the RAI response amended following the audit.

- RAI 841-6055, Question 03.04.01-30: The staff was awaiting the submittal of changes to the building layout, flood barriers, and water-tight doors resulting from seismic design changes.
- Status: The information has been submitted and the staff is reviewing the amended RAI response.

- RAI 546-4345, Question 03.04.02-6: The staff requested MHI to clarify the use of 0.7 as the coefficient of friction at the soil-concrete interface.
- Status: The staff is reviewing MHI's basis for the coefficient of friction as part of seismic review.

Description of Open Items

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- RAI 782-5910, Question 14.03.07-58: MHI included acceptable ITAAC for turbine generator arrangement and turbine missile probability in DCD Revision 2. However, MHI modified the ITAAC in DCD Revision 3 and the staff found them to be unacceptable. Applies to Section 3.5.1.3.
- Status: MHI has proposed modified ITAAC and corresponding Tier 2* information regarding the turbine generator arrangement and turbine missile probability in response to a follow-up RAI. The staff finds the proposed ITAAC and Tier 2* information acceptable and this will be a confirmatory item for DCD Revision 5.

- RAI 758-5680, Question 03.05.03-10: The staff requested MHI to provide an analysis assessing the local effects of an automobile missile on all seismic Category I structures not covered by RG 1.76.
- Status: The staff is reviewing the applicant's response.

- RAI 1013-7031, Question 03.09.02-103: The staff requested MHI to address the postulated steam generator failure mechanisms associated with the San Onofre Nuclear Generating Station with regard to the US-APWR steam generators.
- Status: The staff is awaiting the applicant's response.

Description of Open Items

- RAI 1013-7031, Question 03.09.02-104: The staff requested MHI to provide the preliminary design of the steam generator tube bundle and the design criteria for the steam generator tubes and retainer bars against flow-induced excitations, including random turbulence, fluid elastic instability (out-of-plane and in-plane), and vortex shedding.
- Status: The staff is awaiting the applicant's response.

- RAI 209-1803, Question 03.09.03-21: The staff review of the seismic analysis of major components supports was impacted by MHI's seismic analysis changes.
- Status: The staff found the analysis methods in the MHI's latest technical reports (MUAP-10006, Revision 3 and MUAP-09002, Revision 3) acceptable. Pending the incorporation of MUAP-09002, Revision 3 into the DCD, this issue is a confirmatory item.

- RAI 107-1293, Question 03.09.04-1, Subquestions 1293-01, 1293-06, and 1293-07, and RAI 848-6093, Question 03.09.04-14: There was limited margin between the calculated maximum control rod drive mechanism deflection and the design limit prior to seismic analysis changes.
- Status: The staff is awaiting the updated calculated maximum control rod drive mechanism deflection based on the updated seismic analysis.

Description of Open Items

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- RAI 288-2274, Question 03.09.06-1: MHI has chosen to address implementation of the functional design, qualification, and inservice testing programs in accordance with the requirements of 10 CFR 52.79(a)(11) and therefore will make available for audit a sample of applicable design and procurement specifications.
- Status: Open Item to be closed upon the staff completing and documenting the audit, currently planned for February 2014.

- RAI 486-3861, Question 03.10-17: The staff requested MHI to provide a list of the components of the gas turbine generator system to be qualified and a description of the methods, criteria, and procedures. This information is included in technical report MUAP-10023, which was impacted by MHI seismic analysis changes.
- Status: The staff is reviewing a revised RAI response and a revision to MUAP-10023.

- RAI 650-5093, Question 03.11-39: MHI's commercial-grade dedication process does not provide an alternative means of environmental qualification since it does not address all environmental qualification requirements specified in 10 CFR 50.49 for electrical equipment and the guidelines of ASME QME-1-2007, Appendix QR-B for nonmetallic components of mechanical equipment.
- Status: The staff is reviewing the applicant's response.

Description of Open Items

- RAI 589-4536, Question 03.11-36: The staff has concerns associated with the calculational methods and results for total integrated dose to equipment inside containment following a loss of coolant accident.
- Status: The issue is dependent on the resolution of Chapter 12 open items.

- RAI 589-4536, Question 03.11-37: The staff has concerns associated with the calculational methods and results for the beta ray source term for equipment inside containment following a loss of coolant accident.
- Status: The issue is dependent on the resolution of Chapter 12 open items.

- RAI 589-4536, Question 03.11-38: The staff has concerns associated with inconsistencies in the operability times of post accident equipment inside containment.
- Status: The issue is dependent on the resolution of Chapter 12 open items.

- RAI 880-6142, Question 03.11-42: The staff requested MHI to provide a equipment qualification data package template.
- Status: The equipment qualification data package template provided by MHI is under review.

Description of Open Items

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- RAI 805-5915, Question 03.11-41 and RAI 880-6142, Question 03.11-43: MHI has not demonstrated how the US-APWR satisfies the environmental qualification requirement for electrical equipment (10 CFR 50.49) (1) with regard to the treatment of nonsafety-related electrical equipment located in a harsh environment, whose failure under postulated environmental condition could prevent satisfactory accomplishment of safety functions, and (2) by deleting “Important to safety” words throughout environmental qualification documents.
- Status: The staff is reviewing the applicant’s response.

- RAI 901-6257, Question 03.11-55: The ITAAC do not include demonstration of environmental qualification of nonmetallic parts of mechanical equipment.
- Status: The staff is reviewing the applicant’s response.

- RAI 804-5938, Question 03.12-26: The design loads for piping were updated because of changes to MHI’s seismic analysis methods. In addition, the seismic analysis methods of steam generator supports were unclear.
- Status: Based on a review of MUAP-09002, Revision 3, the staff has found the piping analysis modeling acceptable and the treatment of the steam generator non-linear supports conservative. Pending the incorporation of MUAP-09002, Revision 3 into the DCD, this issue is a confirmatory item.

LUMINANT GENERATION COMPANY

Comanche Peak Nuclear Power Plant, Units 3 and 4

ACRS US-APWR Subcommittee

FSAR Chapter 3 (less 3.7, 3.8) –

Design of Structures, Systems, Components, and Equipment

November 20, 2013

Agenda

- Introduction
- SER Open Items
- SER License Conditions
- Site-Specific Aspects

Introduction

- FSAR uses IBR methodology
- No departures taken from US-APWR DCD
- All COL Items are addressed in FSAR
- 3 SER Open Items
- 5 Chapter 3 SER License Conditions
- No contentions are pending before ASLB

SER Open Items

RAI 244 (6222) Question 3.9.6-21

Provide detail re functional design, qualification and IST programs for pumps, valves and dynamic restraints

Proposed resolution – Typical MHI specs are projected to be available in December 2013

SER Open Items (cont'd)

RAI 239 (6159) Question 3.11-18

Provide detail re implementation of the EQ for mechanical equipment

Proposed resolution –Typical MHI specs are projected to be available in December 2013

SER Open Items (cont'd)

RAI 239 (6159) Question 3.11-19

Provide full description of the Operational EQ Program

Proposed resolution – the operational aspects of the EQ program are addressed in MUAP-08015, which is still under NRC review

SER License Conditions

3-1 Pipe break hazards analysis

Applicant proposed - licensee shall complete the as-designed pipe break hazards analysis before commencing installation of individual piping segments

3-2 MOV testing

Applicant proposed - licensee shall implement MOV testing program prior to initial fuel load

SER License Conditions (cont'd)

3-3 Schedule for inservice testing program

Staff imposed - within 12 months of COL, licensee shall submit schedule for inservice testing program to support planning for and conducting NRC inspection

3-4 EQ Program

Applicant proposed - licensee shall implement EQ program prior to initial fuel load

SER License Conditions (cont'd)

3-5 Schedule for EQ program

Staff imposed - within 12 months of COL, licensee shall submit schedule for EQ program to support planning for and conducting NRC inspection

5-2 Preservice testing

Applicant proposed - licensee shall implement pre-service testing program prior to initial fuel load

Turbine Inspection Program

Applicant proposed – program to be implemented prior to initial fuel load

Site-Specific Aspects

Appendices 3A, 3B, 3C, 3E, 3F, 3G, 3H, 3I, and 3J are incorporated by reference with no supplements or departures

3.1 Conformance with General Design Criteria

ESWS and CCWS piping arranged to permit access for inspection

- **Manholes, hand holes, inspection ports**
- **Integrity of underground piping demonstrated by pressure and functional tests**

3.2 Classification of Structures, Systems, and Components

- Site-specific S-R SSCs designed to withstand the effects of earthquakes without loss of capability to perform their safety function are
  - ESWS
  - UHS (except basin makeup piping and valves)
  - UHS ESWS pump house ventilation
- DCD methods of equipment classification and seismic categorization applied to Table 3.2-201

3.3 Wind and Tornado Loadings

- Site-specific basic wind speed (96 mph) enveloped by standard plant design (155 mph)

- There are no site-specific SC II buildings/structures
  - No site features promoting channeling/buffeting
  - Wind design methods used for standard plant buildings are valid for site

- Site-specific SC I structures (UHSRS, ESWPT, PSFSVs) designed for same tornado loadings and combined tornado effects using same methods for qualification as standard plant SSCs

3.3 Wind and Tornado Loadings (cont'd)

- ❑ **Failure of N-SR buildings/structures will not**
  - **Jeopardize S-R SSCs**
  - **Generate missiles not bounded by FSAR 3.5.1.4**

- ❑ **Site-specific or field-routed S-R SSCs evaluated for structural reinforcement/missile barriers**

3.4 Water Level (Flood) Design

- ❑ **No site-specific flood protection measures applicable since plant is built above DBFL with adequate site grading**

- ❑ **All SC I structures below-grade are protected against effects of flooding, including groundwater**

- ❑ **All site-specific S-R structures (UHSRS, ESWPT, PSFSVs) evaluated for internal flooding**
  - **Each configured with independent compartments, divisionally separated**
  - **Internal flooding of any one compartment and corresponding division will not prevent system from performing required safety functions**

3.5 Missile Protection

- Internally generated missiles from vent fans, pumps and cooling tower fans are not credible
- Units 1 & 2 are outside low-trajectory turbine missile strike zone of Units 3 & 4 and vice-versa
- Unit 3 & 4 T/Gs are in “unfavorable orientation” with respect to S-R SSCs of adjacent US-APWR unit
  - Inspection intervals keep P1 < E-5 per year
  - Therefore, acceptable risk for turbine missiles maintained < E-7 per year

3.5 Missile Protection (cont'd)

- Neither air crash nor air transportation accident is required to be considered due to limited operations per year

- No site-specific hazards for external events produce missiles more energetic than tornado missiles identified for standard plant design

3.6 Protection Against Dynamic Effects Associated with Postulated Rupture of Piping

- No site-specific HE piping is within protective walls of ESWPT and UHSRS
- Site-specific ME piping systems are ESWS and FSS
  - A crack in ME piping does not affect safety function of ESWS and UHS
- Procedures address plant operating and maintenance requirements to prevent water hammer due to voided lines

3.9 Mechanical Systems and Components

- The only site-specific active equipment is UHS transfer system pumps and valves
- IST Program developed using ASME Section XI, ASME OM Code, Tech Specs, and good engineering practice
- IST Program implemented in general conformance with NUREG-1482

3.10 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

- OBE defined as 1/3 SSE
- EQ Program described in MUAP-08015 establishes overall framework for implementation, including seismic qualification
- Plant-specific implementation requires turnover of all EQ Program records to Luminant

3.11 Environmental Qualification of Mechanical and Electrical Equipment

- ❑ Operational EQ Program is established and implemented prior to unit fuel load
- ❑ Equipment test results are maintained as auditable project records
- ❑ Site-specific equipment subject to loss of ventilation is qualified

Luminant US-APWR MNES MITSUBISHI NUCLEAR ENERGY SYSTEMS, INC.

3.12 Piping Design Review

- ❑ Site-specific response spectra is used for analysis of yard piping not in standard plant design
- ❑ ASME III Class 2 or 3 piping is not exposed to wind or tornado loading
  - Exposed non-ASME piping is evaluated
- ❑ There are no exceedances of high frequency CSDRS
  - Screening of piping for high frequency sensitivity not required

Luminant

US-APWR

MNES
MITSUBISHI NUCLEAR ENERGY SYSTEMS, INC.

3.13 Threaded Fasteners (ASME Code Class 1, 2, and 3)

- Threaded fastener Pre-Service Inspection Program will be implemented after start of construction

- Inservice Inspection Program implemented as part of Operational Programs

3D Environmental Qualification List

10 CFR 50 App B criteria applied to seismic category I electrical, mechanical, and I&C equipment in the UHSRS

3K Components Protected from Internal Flooding

Internal flood barrier walls are provided between trains in the UHSRS, ESWPT, and PSFSVs

Acronyms

ASLB	Atomic Safety and Licensing Board
ASME	American Society of Mechanical Engineers
CCWS	component cooling water system
COL	Combined License
CSDRS	certified seismic design response spectra
DBFL	Design basis flood level
EQ	environmental qualification
ESWPT	essential service water pipe tunnel
ESWS	essential service water system
FSAR	Final Safety Analysis Report
HE	high-energy
IBR	incorporated by reference
IST	inservice testing
ME	moderate-energy
MOV	Motor-operated valve

Acronyms (cont'd)

N-SR	Non-safety related
OBE	operating basis earthquake
PSFSV	power source fuel storage vault
RAI	Request for Additional Information
SC I	seismic Category I
SC II	seismic Category II
S-R	Safety-related
SER	Safety Evaluation Report
SSC	systems, structures, and components
SSE	safe-shutdown earthquake
T/G	Turbine generator
UHS	ultimate heat sink
UHSRS	ultimate heat sink-related structures
US-APWR	United States Advanced Pressurized Water Reactor

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Presentation to the ACRS Subcommittee

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

Safety Evaluation Report with Open Items

Chapter 3: Design of Structures, Components, Equipment and Systems

November 20-21, 2013

Staff Review Team

- **Mechanical Engineering Branch**
  - **Yiu Law – Section 3.2.1**
  - **Sardar Ahmed – Section 3.2.2**
  - **Renee Li – Section 3.6.2**
  - **John Wu – Section 3.9.1**
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  - **Tuan Le – Section 3.9.3**
  - **Jason Huang – Sections 3.9.4, 3.9.5**
  - **Thomas Scarbrough – Section 3.9.6, 3.11**
  - **P.Y. Chen – Section 3.10**
  - **James Strnisha – Section 3.13**
  - **Robert Hsu – Section 3.12**
- **Structural Engineering Branch 1**
  - **Sunwoo Park – Sections 3.3.1, 3.3.2, 3.4.2, 3.5.3**

Staff Review Team

- **Balance of Plant and Fire Protection Branch**
  - **Ryan Nolan – Sections 3.5.1.1, 3.5.1.2, 3.5.1.4, 3.5.2**
  - **Raul Hernandez – Section 3.6.1**
- **Materials and Chemical Engineering Branch**
  - **John Honcharik – Section 3.5.1.3**
  - **Eric Reichelt – Sections 3.6.3**
- **Radiation Protection and Accident Consequences Branch**
  - **Rao Tammara – Section 3.5.1.5, 3.5.1.6**
- **Electrical Engineering Branch**
  - **Peter Kang – Section 3.11**

- **Project Managers**
  - **Lead PM: Stephen Monarque**
  - **Chapter PM: Dennis Galvin**

- **Note: Section 3.4.1, 3.7 ,and 3.8 are not addressed in the SE.**

Description of Open Items

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- RAI 244-6222, Question 03.09.06-21: MHI has chosen to address implementation of the functional design, qualification, and inservice testing programs in accordance with the requirements of 10 CFR 52.79(a)(11) and therefore will make available for audit a sample of applicable design and procurement specifications. This open item tracks the issue for Luminant.
- Status: Open Item to be closed upon the staff completing and documenting the audit, currently planned for February 2014.

- RAI 239-6159, Question 03.11-18: MHI has chosen to address implementation of the environmental qualification of mechanical equipment in accordance with the requirements of 10 CFR 52.79(a)(11) and therefore will make available for audit a sample of applicable design and procurement specifications. This open item tracks the issue for Luminant.
- Status: Open Item to be closed upon the staff completing and documenting the audit, currently planned for February 2014.

- RAI 239-6159, Question 03.11-19: The staff cannot complete its evaluation of Luminant's description of its environmental qualification operational program because the review of MUAP-08015, "US-APWR Equipment Environmental Qualification Program," for the DCD is incomplete.
- Status: The review of this open item is on hold pending the completion of the review of MUAP-08015.

Agenda

- Introduction
- SER Open Item
- SER License Condition
- Site-Specific Aspects

Introduction

- FSAR uses IBR methodology
- No departures from US-APWR DCD
- All COL Items addressed in FSAR
- One SER Open Item
- One SER License Condition
- No contentions pending before ASLB

SER Open Item

9.2.5-1 UHS Heat Load

"Governing" UHS heat load for determining basin capacity changed from safe shutdown with LOOP (2-train) to LOCA (2-train). Specific values for heat load and cooling water capacity are expected to change slightly in DCD Subsection 9.2.5.2.3, Tables 9.2.5-1, 9.2.5-2, 9.2.5-3, and Technical Specification 3.7.9

Expected to be resolved in DCD Phase 4

Changes could impact COLA

SER License Condition 9-1

Fire Protection Program Implementation Milestones

- **Storage Buildings for byproduct and special nuclear materials before initial receipt**
- **Areas containing new fuel before receipt**
- **All Fire Protection Program features before initial fuel load**

Site-Specific Aspects

9.1.5 Heavy Load Handling Program

- Program established prior to first fuel load

Site-Specific Aspects (cont'd)

9.2 & 9.5 ESWS and UHS

Will be discussed in Closed Session due to SRI

Site-Specific Aspects (cont'd)

9.4 Ventilation Systems

- Capacities of HVAC equipment reflect site specific conditions
- UHS ESW Pump House Ventilation System maintains proper environmental conditions

Site-Specific Aspects (cont'd)

9.5.1 Fire Protection System

- Program – ensures fire will not affect safe-shutdown capabilities
- Defense-in-depth approach that includes fire detection, extinguishing systems and equipment, administrative controls, procedures, and trained personnel
- Fire Hazards Analysis performed
- Combustibles Control Program

Site-Specific Aspects (cont'd)

9.5.2 Communications Systems

- Various intra-plant and offsite communications systems are provided for normal and emergency communications

9.5.4 Fuel Oil Storage Tanks

- Underground vaults containing fuel oil tanks for Gas Turbine Generators are provided

Acronyms

- ASLB Atomic Safety and Licensing Board
- CCW Component Cooling Water
- COL Combined License
- COLA Combined License Application
- DCD Design Control Document
- ESW Emergency Service Water
- ESWS Emergency Service Water System
- FSAR Final Safety Analysis Report
- HVAC Heating, Ventilation and Air Conditioning
- IBR Incorporated by Reference
- LOCA Loss of coolant accident
- LOOP Loss of offsite power
- NPSH Net positive suction head
- RG Regulatory Guide
- SER Safety Evaluation Report
- UHS Ultimate heat sink
- US-APWR United States Advanced Pressurized Water Reactor

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Presentation to the ACRS Subcommittee

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

Safety Evaluation Report

CHAPTER 9: Auxiliary Systems

November 20-21, 2013

Staff’s Presentation Order

- **Stephen Monarque** - Comanche Peak COLA Lead Project Manager
- **Paul Kallan –** Senior Project Manager
- **Larry Wheeler -** Technical Staff Presenter

Section 9.02.05 Ultimate Heat Sink

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- ***Open Item No. 09.02.05-01***

The "governing" heat load for UHS basin cooling capacity is being changed from Safe Shutdown with LOOP (2-train) mode to LOCA (2-train) mode

- DCD Revision 4 shows a small change in heat load and cooling capacity values
    - Safe Shutdown with LOOP peak heat load remains the same at 196 E^6 BTU/H/train
    - LOCA peak heat load change from 158 E^6 BTU/H/train to 161 E^6 BTU/H/train (~2% increase)
    - LOCA 30 day heat load for UHS slightly affects 30 day cooling tower capacity

- These changes are expected to impact COL FSAR 9.2.5.2.3:
    - **Total cooling water capacity (gallons) – from 8.4 E^6 to 8.3 E^6 (Safe Shutdown with LOOP)**
    - **Total cooling water capacity (gallons) – from 8.2 E^6 to 8.4 E^6 (LOCA)**

- **Concerns: NONE – total UHS basin water volume remains the same**

Acronyms

- COL – combined license
- COLA – combined license application
- DBA – design basis accident
- FSAR – Final Safety Analysis Report
- GDC – General Design Criteria
- IBR – incorporated by reference
- SER – Safety Evaluation Report
- RAI – request for additional information
- RCOL – reference combined license