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UNITED STATES NUCLEAR REGULATORY COMMISSION'S

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

June 6, 2008

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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553RD MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

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FRIDAY,

JUNE 6, 2008

The Advisory Committee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B3, 11545 Rockville Pike, Rockville, Maryland, at 8:30 a.m., WILLIAM J. SHACK, Ph.D., Chair, presiding.

MEMBERS PRESENT:

WILLIAM J. SHACK, Chair
MARIO V. BONACA, Vice Chair
JOHN D. SIEBER, Member-at-Large
SANJOY BANERJEE
J. SAM ARMIJO
DANA A. POWERS
SAID ABDEL-KHALIK
OTTO L. MAYNARD
JOHN W. STETKAR
DENNIS C. BLEY
MICHAEL L. CORRADINI
GEORGE E. APOSTOLAKIS

NRC STAFF PRESENT:

JEFF CIOCCO
MICHAEL SCOTT
BILL RULAND
PAUL KLEIN
JOHN GROBE

ALSO PRESENT:

KIYOSHI YAMAUCHI
KEITH PAULSON
SHINJI KAWANAGO
HIROSHI HAMAMOTO
MASAYA HOSHI
TADASHI SHIRAISHI
MUTSUMI ISHIDA
KATSUNORI KAWAI
MAKOTO TAKASHIMA

I-N-D-E-X

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

14) OPENING REMARKS BY THE ACRS CHAIRMAN

CHAIRMAN SHACK: The meeting will now come to order. This is the third day of the 553rd meeting of the Advisory Committee on Reactor Safeguards. During today's meeting, the Committee will consider the following: an overview of the U.S. advanced pressurized water reactor design and the status of NRC activities associated with the resolution of generic safety issue 191, "Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance."

This meeting is being conducted in accordance with the provisions of the Federal Advisory Committee Act. Mr. Tanny Santos is the designated federal official for the initial portion of the meeting.

We have received no written comments or requests for time to make oral statements from members of the public regarding today's session. We have representatives of the State of Vermont on the phone bridge line listening for discussion of the topics scheduled for today's meeting.

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

A transcript of portions of the meeting is being kept. And it is requested that speakers use one of the microphones, identify themselves, and speak with sufficient clarity and volume so that they can be readily heard.

I would remind the members that we are scheduled to interview another candidate today at the end of the presentations. And so don't disappear.

With that, I will turn the meeting over to Otto, who will lead us through the first presentations.

MEMBER MAYNARD: Thank you, Mr. Chairman.

15) OVERVIEW OF THE US-ADVANCED PRESSURIZED WATER REACTOR DESIGN

15.1) REMARKS BY THE SUBCOMMITTEE CHAIRMAN

MEMBER MAYNARD: As you said, this is a brief overview of the U.S. APWR. It's an informational meeting. We are not expected to write a letter or to make any decisions. So we don't need to get into the level of detail that we might need to thoroughly examine every aspect of this.

So we have a lot of material to cover. I think you are going to find that this is a very complete package on the information. It covers comparisons to U.S. current plants. It provides numbers, good diagrams, fuel design. It identifies the various codes and methods that they're using for various parts of the analysis, safety analysis; fuel design containment. It provides some of their information relative to GSI-191.

It also provides some good information on some of the unique aspects of this. It has an advanced accumulator that is a little different than what we have seen. It has digital I&C control room systems that I think some really good information is going to provide about that.

The reason I am identifying all of this is that we will hold most of our questions toward the end. I think that we will find that the presentation will cover the items and then maybe have a little bit of time at the end for some discussion on some of the items. So I will be trying to control that as we go through.

(Laughter.)

MEMBER POWERS: Good luck.

MEMBER MAYNARD: I know. This plant, one of the plants that they're comparing it to, it looks to me from the numbers, is one of the later models. Westinghouse PWR is a SNUPS design. It looks like some of the Calloway/Wolf Creek numbers that I see in there as part of the comparison there. So it's kind of near and dear to my heart there.

With that, I want to turn it over to Jeff Ciocco, the staff. He will lead the presentation. And then I think Mr. Kiyoshi Yamauchi will be leading the Mitsubishi presentation. So I will turn it over to Jeff to get started.

MR. CIOCCO: Okay. Thank you.

15.2) BRIEFING BY AND DISCUSSIONS WITH REPRESENTATIVES OF THE NRC STAFF AND MITSUBISHI HEAVY INDUSTRIES

MR. CIOCCO: My name is Joe Ciocco. I am the lead project manager for the U.S. APWR standard design certification. I am going to give you a brief project overview this morning before Mitsubishi gets into their technical presentation on their reactor design technology.

So the purpose of this morning's briefing is to provide information to you to familiarize you with the U.S. APWR design certification application, the licensing review process, and the current status of our licensing review and to address any questions that you have.

So the agenda is going to be short. I will talk about the application status, the review schedule, and the information provided in the design control document that we have, as well as the topical reports.

The U.S. APWR application status. We began interactions with Mitsubishi back in July of 2006. These were called the pre-application review meetings. We had ten meetings with Mitsubishi. They came to us in early 2006 with their intent to apply for a standard design certification.

So we had ten public meetings, pre-application review meetings. And most of these were meetings. They were pre-submittal meetings of topical reports that Mitsubishi came to us and said they had 12 areas that they wanted to supply topical reports. And prior to those topical report submissions, they wanted to have a public meeting with us.

They chose the areas of the topical reports, the accident analysis, digital I&C, the advanced accumulator, thermal design methodology, and fuel design methodology, as well as the quality assurance program description.

So from July 2006 until the tendering of the application on December 31st, 2007, we had pre-application review meetings with Mitsubishi.

Their application was tendered to us on December 31st, New Year's Eve, of last year. We completed a 60-day acceptance review and docketed the application on February 29th. We have a docket number.

So right now we are about three months into our phase 1 licensing review. The staff is writing RAIs, requests for additional information; as well as writing the preliminary safety evaluation report. So we are really in the early stages right now of phase 1 of six phases to produce the final safety evaluation report. And I will show you a schedule in a few slides.

I wanted to put on here as far as our -- you've heard DCWG, design-centered review groups, where we have the design technology and the utility who selected that. In this case we have Luminant Power, which selected the U.S. APWR technology for their units 3 and 4 at the Comanche Peak site.

MEMBER SIEBER: It would be good if you could provide us with a disk with the DC as submitted on it.

MR. CIOCCO: I certainly will. It's also --

MEMBER SIEBER: Give us a head's up and ability to familiarize ourselves with the plant before we start on the iterations.

MR. CIOCCO: Certainly will. We also have it on our public Web site as well.

MEMBER SIEBER: I would prefer a disk.

(Laughter.)

MR. CIOCCO: We can do that. I have them at my desk.

MEMBER SIEBER: I know it's on the Web site.

MR. CIOCCO: Okay. A little bit on the review schedule background. This is very important for us in building our schedule for the licensing review.

Like I said, we received the design control document. It defines a very specific approach where there is design criteria and there is a process for Mitsubishi submitting technical reports to us through its what they call a design timeline or very particular areas, for instance, in chapter 3 in their piping and components and the digital I&C.

The application references the 13 topical reports. One of those is completed now where the staff has written the safety evaluation report. And that's on the quality assurance program description.

We have 12 topical reports currently under staff review. And they are all referenced in the design control document. And there are 15 technical reports, which aren't stand-alone documents, but they are a supplement to the design control document that the staff is reviewing along with a particular chapter of the DCD.

And of these 50 technical reports, about 25 have been tendered to the NRC. And there are about 25 more to come in. The last ones come in -- right now the schedule is -- mid 2009. A lot of them are the stress analysis results of the piping and components and I think a seismic analysis of the fuel design.

So we are well on our way over. I would say 25 or 30 of the technical reports have been submitted to the NRC. MHI had a goal that was to minimize the scope, a number of the open items, if you will, at the end of phase 2. That is really our draft safety evaluation report. We call it the SE with open items.

So whenever we built our schedule, we looked at, when was the staff going to complete its review of the topical reports and when are we going to have all the technical reports in for review so the staff can make an evaluation finding?

If necessary, at the end, we are going to re-base on our schedule at the completion of phase 2 in the ACRS when we know the number and complexity of the open items that we're going to have. We will certainly coordinate with the ACRS staff your review twice, what we call phases 3 and 5. Three is the SE with open items. And then phase 5 is the draft final safety evaluation report.

MEMBER MAYNARD: And, for the Committee's information, I talked with Jeff a little bit. We're going to need to be taking a look at the list of topical reports and identifying what items that we may want to review before we see the SER with open items and stuff, too.

MEMBER ARMIJO: Will this list of topical reports include the various analytical codes used in the safety evaluation? Are you going to review these on a generic basis or is it going to be kind of mixed in with the licensing or certification of the plant?

MR. CIOCCO: The topical reports do include computer codes and the thermal hydraulics for the advanced accumulator and the accident analysis. So the staff is currently reviewing those in the topical reports.

MEMBER ARMIJO: For example, fuel design

methods?

MR. CIOCCO: Yes.

MEMBER ARMIJO: That will be included if it isn't already?

MR. CIOCCO: It is in a topical report. And you will see in Mitsubishi's presentation where they are going to talk about a particular area in the topical report and the staff, where we are doing the review.

Our last topical report came in I think in March of this year. It's on a particular code. It's a Mitsubishi code called FINDS, F-I-N-D-S, which is used in their fuel design. So that's currently under staff review.

MEMBER SIEBER: So are we going to get the opportunity to review the topical reports along with the application?

MR. CIOCCO: Yes. The HRS has each and every topical report that's been tendered to the NRC. And I'm going to show you a slide shortly which is going to show the expected completion date of the NRC review. And you will see that those dates are actually very close to the end of the staff's review for a particular chapter.

So you will be able to look at the staff's safety evaluation report on that topical as well as the staff's safety evaluation report on the chapter of the DCD which references that topical report.

MEMBER MAYNARD: I'm going to say probably at our next meeting, we need to go over a list, identify what things we have an interest in, and make sure that we get involved at a time that it will do some good.

MR. CIOCCO: Okay. Next I show this is the overall review schedule. We put this in a letter to Mitsubishi on May 9th. It shows about a 42-month review schedule from the phase I.

The application was tendered, like I said, on December 31st. We docketed in February. Our start date was around the middle of March. So it's going to take us about 24 months to get to our safety evaluation report with open items.

At that point -- and this is the last date of a chapter completion. There are 19 chapters of a design control document. So we have a schedule leading down to March 10th, when all of these chapters are going to be completed. We will look at interacting with the ACRS subcommittee on where they are going to receive those chapters.

The overall schedule down through the

final safety evaluation report with no open items is about a 42-month schedule. This will be followed through a rulemaking process, like we did for all other standard design certification.

MEMBER CORRADINI: Let me ask you a question at this point. What did you learn from the ESBWR and the AP1000 that has improved this schedule or made it better for the staff's interaction with ACRS and with the applicant? Is anything different here than the ESBWR schedule, for example?

MEMBER CORRADINI: Well, from a scheduling standpoint. I am trying to understand the differences or is it following the ESBWR schedule?

MR. BURKHART: I am Larry Burkhart of the U.S. APWR, projects branch chief.

We have learned a lot from ESBWR, and we think it is an advantage for them to come to you with specific chapters. I hope you think that is an advantage, too. And I see some mixed looks.

We are open to discussing how we do that, but we are starting, Jeff and I are starting, to discuss how and when we are going to come to you. And, of course, we will work with your staff on when that might be, the best time to do with SER with open items.

And it looks like the chapters may be finishing up in two waves, and they make sense once we get into the scheduling and come to you with specific --

MEMBER POWERS: Let me say that you choose your waves, at least with respect to the EPR, where I know what the waves will look like, is curious. You bring a wave forward and say, "Okay. We're going to look at the i&c systems before we look at the plant layout and whatnot. That is not going to work.

MR. BURKHART: We would definitely take your input on that and do it in a way that makes sense.

MEMBER MAYNARD: I think it's fair to say that the ACRS has learned some lessons from the ESBWR review. And we will have some interactions and coming up with what we think the optimum --

MR. BURKHART: Absolutely. And we have been working with Tanny. And we will take your input and try to make that more efficient.

Also, we may want to come to you on particular issues as you discuss like unique design features of the advanced accumulator before the SER with open items. And we are definitely open to doing that. I think that's a good idea.

MEMBER CORRADINI: So I had a second question, which is let's take the March 2010 date. That's the end of the 20-19 chapters with SER open items or that's when the staff will start feeding it in some fashion to the ACRS?

MR. CIOCCO: This is the end date of the total completion of the compilation of the safety evaluation report of all 19 chapters into the final document, if you will, of an SE with open items. There are individual completion dates leading down to this. If you were to look at a game chart, as we heard --

MEMBER CORRADINI: I don't want to, but I think I've got you now. So that's the end game. So is it your anticipation that with some sort of wave structure, that a subcommittee of ACRS will see groupings of the chapters before March 10th?

MR. CIOCCO: Yes, sir.

MEMBER CORRADINI: Okay.

MR. CIOCCO: Yes.

MR. BURKHART: Yes. Those are not-to-exceed dates. And as we were thinking of how phase 2 is going to wind up, we hope to come to you with some chapters, whatever makes sense, before that March 2010 date exactly.

MEMBER CORRADINI: Thank you.

MR. BURKHART: You are welcome.

MR. CIOCCO: Okay. Next I have two slides which show the 19 chapters of the application that we have in each of our -- we have a project team in the Office of Nuclear Reactors as well as the topical reports.

As I said, there were 13 topical reports. And in parentheses, I put the completion date of these topical reports. And, like I said, I think the ACRS has each of these. We can always provide them again and then the expected dates.

And the topical reports are the areas that Mitsubishi has chosen to work with the staff. In the area of the reactor, chapter 4, you have the fuel design, thermal design methodology. And this was our latest topical report. The May 2009 is the completion date for the fuel assembly.

In chapter 6, we have the advanced accumulator you're going to hear about as well as the LOCA mass and energy. Chapter 7, digital I&C has three topical reports. As you said, each of these chapters has a particular completion date leading up to that March 2010 date.

We have the accident analysis,

large-break, small-break, and non-LOCA analysis. This one is completed on the quality assurance program description. And, then, finally, we have the human factors and HSI system description.

So that is really how we laid out our project assignments and topical report assignments because these will be stand-alone safety evaluation reports. And in many of these, Mitsubishi has asked that these topical reports be applied to the operating fleet as well as the new reactor fleet.

MEMBER BLEY: Question on that last item you have. Chapter 19 doesn't seem to have any associated reports with it. Is there a PRA done yet?

MR. CIOCCO: Oh, yes, sir, there is. Yes. They have actually submitted a PRA level 3.

MEMBER BLEY: So you do have --

MR. CIOCCO: Yes. They just don't have a topical report. But it was totally addressed in the chapter 19 document.

MEMBER STETKAR: Do they have the actual PRA?

MR. CIOCCO: Yes, sir.

MEMBER BLEY: You have it?

MR. CIOCCO: We do. Yes, sir, we do. Yes.

MEMBER BANERJEE: So do you have a schedule for when you complete the SERs for the topical reports?

MR. CIOCCO: Yes, sir. These are the completion dates in parentheses.

MEMBER BANERJEE: Oh, okay.

MR. CIOCCO: Most of them, they were tendered beginning in January of 2007, the advanced accumulator, and the quality assurance program description. So they have been coming in in kind of a steady stream beginning in early 2007 through the last and final receipt was in March of this year. These are the completion dates that I put in parentheses.

MEMBER BANERJEE: So we should actually schedule some subcommittee meetings if you wish to review these? And those dates are the ones when you finish your SER?

MR. CIOCCO: This is when the staff is going to have its position on those topical reports.

MEMBER MAYNARD: I think we are going to have to get -- let's not SER with open items. Let's --

MR. CIOCCO: These are stand-alone safety evaluation reports on the top row.

MEMBER MAYNARD: We need to get involved or that's what we need to be identifying.

MEMBER BANERJEE: We need to identify which ones and have the appropriate subcommittee --

MEMBER APOSTOLAKIS: So we get involved after these dates, right?

MEMBER MAYNARD: No. For the topical reports, we're going to need to get involved before or the ones that we choose to take a look at.

MEMBER APOSTOLAKIS: Okay.

MR. BURKHART: Just to make a comment. Because we have less time in pre-application with this applicant than some others, a lot of the topical reports, which are generally their approaches to design, are simultaneous now with the chapters for the DCD.

So whether or not the timing is good to separate those, we're open to hearing that from you, but it's a good opportunity I think to wrap them almost together. But we can talk about that because we can see the advantage of how we're proceeding here is we see that approach and we see how they're implementing their approach.

MEMBER MAYNARD: The situation we don't want to get into is to be reviewing a chapter and a conclusion is drawn on a topical report we didn't get a chance to take a look at that, all of a sudden, that leaves us with nothing that we can really comment on or do.

MR. BURKHART: And we'll definitely --

MEMBER MAYNARD: We'll take a look that for - we need to move on here.

MEMBER CORRADINI: Mr. Chairman, may I ask one clarification? So you made a comment in the middle of this. Besides new plants, these topical reports are for, and I didn't get your -- so what plants, currently operating plants, are these topical reports going to be applicable to? That's what I thought you were implying. Maybe I misunderstood.

MR. CIOCCO: Whenever they tendered the topical reports, they asked if these could be applied and used by the operating fleet as well as the new reactor fleet.

MEMBER ARMIJO: And that is for approval of the topical report, --

MR. CIOCCO: Yes, sir.

MEMBER ARMIJO: -- not just for the APWR?

MR. CIOCCO: Not just for the APWR.

MEMBER ARMIJO: All right. Thank you.

MR. CIOCCO: Yes. You're welcome.

MR. BURKHART: And that is a good point because NRR -- we would like them to review it with us. And they're coming back to us and saying, "Well, no one has referenced this in the license amendment," et cetera. So they don't know how to prioritize that.

But the bottom line is we, NRO, are looking at the review. We are coordinating it with NRR.

MEMBER ARMIJO: Well, so --

MEMBER MAYNARD: We do need to move on.

MR. CIOCCO: Yes.

MEMBER MAYNARD: We have a lot of information. This is --

MR. CIOCCO: Okay. Thank you. And that concludes my presentation. I wanted to give you an idea of the application, let you know that the phase 1 is underway with the topicals, technicals, and the application and that we look forward to working with the ACRS as we complete our safety review.

MEMBER MAYNARD: Okay. Thank you.

I think, Mr. Yamauchi, we will get your slides up here on the --

MEMBER BLEY: While we are getting ready here, will this talk tell us about what makes this the U.S., as opposed to others? We think you are building one of these in Japan already.

MEMBER MAYNARD: That is covered, yes.

MR. YAMAUCHI: So, good morning, ladies and gentlemen. My name is Kiyoshi Yamauchi, Executive Officer of Mitsubishi Heavy Industries. This is a great honor for us to have this opportunity to talk with our U.S. APWR, with the ACRS Chairman, and the members. We have many colleagues. Half of them are working in Arlington, half of them from Japan.

I would like to explain very briefly what is MHI. I will talk about experience, technologies, and commitment.

Next, please. Our history in the nuclear world is quite long. And we have been operating already 26 old PWR plants in Japan, including one nuclear ship, MUTSU, at the very beginning. There are 23 plants now in operation, and one is under construction. This is the Tomari unit, this one. And here we will install all digital I&C systems. And they are already there.

We have two plants under licensing, these two. This is APWR plants. So this is the best plant over U.S. APWR and now is under licensing stage.

Even in the stagnation period in the world, we have been continuing to build or replace plants. And that is why we can develop our own technologies as core competence. And also we can keep our engineers and infrastructures.

Next, please. And also we have extensive experiences of components exports. These are for the fourth, steam generators on the vessel head. And if you stop over at Kobayashi shipyard in Japan, you can see a steam generator for San Onofre, big one, and steam generator for EDF in Belgium. And also you can see a big vessel for Okiluoto, EPR, in Finland. Please be visiting there.

Next, please. And the other talk about our technologies. We are not only a manufacturer/vendor. We are a total plant maker. We are a single point responsibility, single turnkey work from the R&D design and engineering, manufacturing, construction, maintenance, and fuel supply.

And also we have established our global assurance during the export phase of components. And this is why we have decided to have the DCD of our own plant. This is why we are here.

Next, please. And one of our core competencies is nuclear safety analysis and with the core designs. We use state-of-the-art technologies. And also we have our own test facilities. And also we will do licensing support to the governments.

MEMBER BANERJEE: Is that in Takasago or where is that?

MR. YAMAUCHI: Yes, yes. Sure.

Next, please.

MEMBER BANERJEE: Full height?

MR. YAMAUCHI: It depends on the -- if full height is needed, we will do that.

Next, please. And also we have a plant engineering and the purchase procurement capability. We will make our 3D-CAD, which integrates common database from design to purchase and construction. It's a once-through system.

Next page, please. And also we have a capability of manufacturing. This is a picture of Super Miller reactor vessel. We can make reactor vessels, steam generators, clDM. We have internals or turbines or all components. We can do that.

Next, please. And also we will do plant constructions ourselves. And the most important issue at this stage is how to shorten the construction period. And the left one is the latest plant in the Hokkaido area. We will use super wash, cranes. And share portion and the dome portion are welded at the site. And they can fix it.

And also we can use a modular utilization effort. And that can enable to reduce the construction period. And we would like to use such experience in U.S., too.

Next page, please. And also we have been supplying fuel assemblies. And so far we have supplied around 18,000 fuel assemblies to Japan PWR plants.

Next, please. And we think, MHI thinks, safety is the most key important issue for the nuclear facilities to develop, to construct, and to operate. So our U.S. APWR is totally in compliance with the U.S. requirements.

And also our U.S. APWR design is very evolutionary type, not revolutionary, not surprising for you. And it's quite similar to this type. But capacity is larger because we can use larger turbines and larger steam generators, which we can make.

And also the systems are quite similar to the conventional ones. And we will use some unique systems. They are all already proven or accepted technologies having used.

The last, please. And there is a conclusion. We are committed to provide the highest quality global nuclear products and services and also our infrastructure, having developed and maintained throughout our long history. And the U.S. APWR demonstrates the commitment to quality and safety worldwide.

This is my short statement. Thank you very much.

MEMBER MAYNARD: Thank you.

MR. YAMAUCHI: Also we have many people. So please do not hesitate to ask many questions.

(Laughter.)

MR. YAMAUCHI: Thank you very much.

MEMBER MAYNARD: We are going to be discussing more on fuel and stuff here when they go through the specific design. This is not the end of the presentation. We need to be moving along here. Keith, are you going to be --

MR. PAULSON: Yes, yes.

MEMBER MAYNARD: -- now going through the design features. And that is where we are going to talk more about some of the specifics on the fuel and stuff. So go ahead.

MR. PAULSON: What I am going to try to do is to supplement some of the things that Mr. Yamauchi started with to give you an identification of how the design looks compared to designs you have seen already.

As he mentioned, you will see very quickly that this is not a revolutionary design but an evolutionary design. It's consistent with designs that have been implemented in the U.S. in many cases and also consistent with the APWR and Japanese designs because of the consistency of the designs manufactured by Mitsubishi with designs that you have seen from the U.S.

So you will see a few new things, which hopefully you will ask some questions on at the appropriate time. And I'll leave that up to Mr. Maynard to make that decision when the appropriate time is. But, in any case, I will move forward.

I am going to have to go through things very quickly because I know you want to leave some time at the end for questions. I have a lot of material. I am not going to go over things you have seen already with a lot of detail. I am just going to point to the fact that it is consistent with something you have already seen.

If there's a question about that, fine. But I am going to try and spend most of the time on those things that you have either asked questions on already or things that may be somewhat new or more advanced that you have seen to date. So that's going to be my philosophy.

These are the issues we're dealing with. And I'm going to deal with what the U.S. APWR is, core and fuel design, systems design. We will talk a little bit about the I&C architecture and also just some conclusions at the end very briefly.

This is going to be really the world's largest, at least we think the world's largest, PWR based on the fact that we'll be getting about 1,700 megawatts electric out of the plant.

One of the reasons that this plant can meet that high objective is because of the high thermal efficiency. And that comes about because of the design of the turbine that we're using, which will have about a 39 percent efficiency, as opposed to typical plants today, which are more in the range of 35 percent.

So we also have a very high -- and this may be of some interest when you look at the steam generator design if you're so inclined to see the high performance of the separators on this design, very, very high-level efficiency on the separators. We are developing the capability to utilize a 70-inch class blade for the turbine, which allows us to get the 39 percent thermal efficiency.

And the U.S. APWR does meet what we believe to be the U.S. utility requirements that have been specified years ago in the utilities requirements document.

MEMBER BANERJEE: Is there any reheat or is it --

MR. PAULSON: Yes.

MEMBER BANERJEE: There is reheat?

MR. PAULSON: Yes. Some of the URD areas that we hit upon specifically and some that are specifically focused on safety, first of all, we have eliminated the penetrations of the vessel on the bottom.

We have implemented full four-train safety, both in terms of the mechanical and electrical components. We are utilizing 14-foot fuel. We have a fully digital I&C.

And I am going to spend more time, obviously, on all of these issues, but this is just some highlights up front.

And due consideration against protection against airplane crash and long-term containment integrity have been built into the design or are being built into the design based on evaluations of things like the airplane crash.

You had some questions about the APWR. This is a generation, more or less the next generation of the APWR, which is going into the Tsuruga 3 and 4 units in Japan. There have been developments that have gone beyond that, and I will talk about those. And you will see those in the comparison tables.

Here shows you some of the testing that went on and the key areas that were identified specifically to demonstrate the acceptability and adequacy of the APWR design. That is in the area of reactor internals and the neutron reflector.

By the way, we have done some confirmatory testing on those again, in addition to what we had done for the APWR. And that was done last year and early into this year.

The compact steam generator design. This looks at a triangular pitch, where we have been able to reduce the pitch on the steam generator tubes and actually reduce the size.

As I mentioned already, the separator performs extremely well for the steam generators and has demonstrated a very low moisture carryover.

The reactor coolant pumps are the same reactor coolant pumps that we would be using for the

APWR design, will be used in the U.S. APWR design. The i&c is an architecture that utilizes complete digital I&C for both the control and protection system and has a history that I will go through later on to show where we started out with respect to this design to show that we just didn't jump right into a design that we plan on implementing here in the United States that is one that has been sequentially developed as backfits for the U.S. or in Japan and will be put in also to the U.S. APWR but has been installed already, by the way, at the Tomari site.

MEMBER SIEBER: What kind of tube support plates do you have in the steam generator?

MR. PAULSON: Tube support plates are stainless steel broached.

MEMBER SIEBER: Broached? Thank you.

MR. PAULSON: And the turbine I mentioned, we are looking at turbine performance. We also will do some additional testing on the new turbine blade.

A quick comparison here. This is one of the things you asked for early on to get to where we're at. We like to look at the U.S. design of the current four-loop plant. This is very similar to the SNUPS design.

This is the APWR as configured in the one that will go in at Tsuruga 3 and 4. And this last slide shows the comparison with the U.S. APWR.

There is one critical key point here, and that is that notice that the thermal output for the U.S. APWR is no larger than that already in the APWR. The only way that the additional megawatts electric were able to get to that is with the higher efficiencies that are identified as part of the design.

So we're not bumping this thing up in power, thermal power, just to get additional megawatts out. We are actually using the performance of the turbine to get those additional megawatts.

MEMBER CORRADINI: And that is all due to the final low-stage blade that we're --

MR. PAULSON: Primarily, yes, you're right. That's the --

MEMBER BANERJEE: Is there any 70-inch blade in operation?

MR. PAULSON: No. That's why I said there will be some additional testing.

MEMBER BANERJEE: And you are doing testing of erosion and all that sort of stuff?

MR. PAULSON: Right. That will be correct. We are doing testing.

MEMBER BLEY: Do you have a full-scale model?

MR. PAULSON: Yes. We will, yes.

MEMBER BANERJEE: It has to be close to Mach 1, the blade tip velocities. Is it 1,800 rpm turbine?

MR. PAULSON: Eighteen hundred rpm.

MEMBER SIEBER: And these are add-on blade rows, as opposed to --

MR. PAULSON: Longer blades, right.

MEMBER SIEBER: So same number of --

MR. PAULSON: Same number of blade rows. I think that's correct. Same number of blade rows. Just the blade is longer.

MEMBER SIEBER: Or more blades?

MR. PAULSON: No. I think it's the longer blade. It goes from 54 to 70 inches, roughly 70 inches.

MEMBER ABDEL-KHALIK: You refer to the steam generators as compact. How does the water inventory in the steam generator compare to that in the 54F model?

MR. PAULSON: There is more. There is more in the steam generator than in the 54. And you can see also just you could guess at that just by the

-- well, let me go to this. This is more or less the square foot area of each of the steam generators. So the 54 by 54 thousand, this has about 91 thousand.

MEMBER ABDEL-KHALIK: Well, I'm not concerned about --

MR. PAULSON: I know, but you asked additional water. Yes, it has more water volume also.

MEMBER BANERJEE: But it is on a triangular pitch, right?

MR. PAULSON: It is on a triangular pitch, right.

MEMBER BLEY: Why do you call it compact?

MR. PAULSON: We call it compact because we reduced the tube size and were able to reduce the size based on what we had used in other designs.

MEMBER BANERJEE: Is the tube sheet the same size as the 70F or is it larger?

MR. PAULSON: No. This is a bigger steam generator.

MEMBER MAYNARD: We will have an opportunity to go into more detail when we go through our other reviews. We need to move on. If we have time at the end, we can have additional discussion on this.

MR. PAULSON: Okay. I have covered most

of these points, I think. The bigger output is not due to additional core power. It's due to just thermal power or the capability of the turbines. And the high-performance turbine is key to getting those additional megawatts.

A few more comparisons. Notice about 193 assemblies, 17 by 17. It's a 17 by 17 you have known and loved in the past or known and hated depending on whether you like it or not. But 17 by 17 is the standard fuel design.

And it's consistent among all three plants. The only difference between the U.S. APWR and the current design is that the U.S. APWR will be a 14-foot core, as opposed to a 12-foot core.

The reactor vessel internals is slightly different because we are not using this baffle/former design. We are using a neutron reflector.

If you remember, I did show a number of tests that went on back in the 1990s on the neutron reflector. We have also done some testing this year that went into this year on the neutron reflector to validate some of the test information that we had.

This is very simple, by the way. And I will get into it a little more later on, this neutron reflector.

The in-core instrumentation was bottom-mounted in the current U.S. designs, was going to be and will be bottom-mounted in the APWR, but we have gone to a top mount for the U.S. APWR.

MEMBER SIEBER: That slows down your fueling a lot, right?

MR. PAULSON: Pardon?

MEMBER SIEBER: Top mount slows refueling?

MR. PAULSON: A little bit, yes. You have more there but not much. But it does allow you to go to 24 months. This design goes to 24 months. So that helps cut the --

MEMBER BANERJEE: That is why you made 14 foot, right?

MR. PAULSON: Right.

MEMBER BANERJEE: So it's not a local limit or a DND limit.

MR. PAULSON: The power stays the same. And what you are going to see --

MEMBER BANERJEE: It has more fuel?

MR. PAULSON: Well, more fuel, yes. Longer fuel, the same number of assemblies, though, as the APWR.

MEMBER BANERJEE: Just to give you a little summary of the types of systems that we're using, we're using, rather than the two-train design for the electrical design for its standard in the U.S., four-loop design and that would be in the APWR, we have gone to four electrical trains. And as we are going in the APWR, we are going to four trains, mechanical. We will have four mechanical trains also in the U.S. APWR.

From a systems point of view, one of the changes we are making here is we are going to -- rather than using two high-head pumps and two low-head pumps, we are going to using four high-head pumps plus the utilization of the advanced accumulator in the U.S. APWR. I will have some more slides on that later on. It's just what the advanced accumulator looks like and how it performs. But, in any case, what it does basically, it allows us to eliminate the low-head pumps on the ECCS design.

Rather than using an outside containment or an outside pit for refueling water storage, we go to the -- that pit is located on the inside for both the APWR and the U.S. APWR. Containment vessel in each case is pre-stressed concrete. And the I&C here, as you can see, will be full digital in the APWR and the U.S. APWR. And, as I mentioned before, that design has already gone into Tomari, which is not an

APWR.

MEMBER MAYNARD: Make sure everybody caught one of the key things here. Instead of 2 100 percent trains for a number of these, there are 4 50 percent trains.

MEMBER STETKAR: I wanted to ask about that key thing. Is that 4 50 percent in licensing space or in functional space? In other words, do I really need two high-head safety injection pumps to meet the thermal hydraulic requirements for, let's say, a small or medium LOCA?

MR. PAULSON: No. You have the capability here of since you have four of them, you can have a single failure and you can have one out of service and still meet LOCA.

MEMBER STETKAR: That's licensing space.

MR. PAULSON: Right.

MEMBER STETKAR: I'm asking, will one pump deliver enough flow? In the 2 by 100, obviously one pump will deliver enough flow for any accident conditions.

MR. PAULSON: Right.

MEMBER STETKAR: Will one of these pumps also deliver enough flow for those same accident conditions?

MEMBER SIEBER: For a small to moderate break size, yes.

MEMBER STETKAR: I am asking --

MR. PAULSON: For certain break sizes, yes. I think for the limiting break, maybe not. I don't know if we --

MEMBER MAYNARD: We will have an opportunity to explore that in far more depth. I want to make sure everybody caught that.

MEMBER BANERJEE: What about long-term cooling? How do you do that?

MR. PAULSON: Long-term cooling? Well, long-term cooling is done with the high-head pumps. And, of course, we have RHR pumps available, too, later on.

The RHR pumps are now used jointly as containment spray. You will see that in some of our slides. But just as a forerunner of that comment, the RHR system is used as a dual function.

MEMBER ABDEL-KHALIK: What is the shutoff head of the high-head pumps?

MR. HAMAMOTO: This is Hiroshi Hamamoto.

It depends on the sheet. After LOCA, only the higher --

MR. PAULSON: Just the high-end.

MEMBER ABDEL-KHALIK: This is a specific question about the pump. What is the shutoff head of the pump?

MR. HAMAMOTO: About 4,000 feet for the high-end pump.

MEMBER ABDEL-KHALIK: Four thousand feet. Translate that to psi.

MEMBER BANERJEE: Thirty feet is 15 psi.

MEMBER ABDEL-KHALIK: So that is really an intermediate pressure pump.

MR. PAULSON: Right. What is it, about 1,300?

MEMBER BANERJEE: It is below saturation for the system or is it above saturation for the system?

MEMBER ABDEL-KHALIK: It doesn't make sense.

MEMBER BANERJEE: Yes. We need some numbers.

MEMBER ABDEL-KHALIK: That doesn't make sense.

MEMBER MAYNARD: Maybe we can get that later.

MEMBER BANERJEE: Can we make a list of things --

MEMBER MAYNARD: We will certainly make a -- yes.

MR. PAULSON: The analysis and the description of all of the systems are part of the design that was submitted. I&C, fully digital in both cases.

We will move on. Fuel assembly. Fuel assembly is fairly standard. We're using a 14-foot with 11 grid. The difference here is that there are 11 grids, as opposed to 9 in some of the current designs. And so, therefore, the distance between them is less than or equal to what you are seeing in current designs.

I think that is the only big difference between other 14-foot 17 by 17s that you have seen using Zirlo for the fuel, zircalloy for certain aspects of the design also.

MEMBER BANERJEE: Did you have full-scale critical heat flux on this testing?

MR. PAULSON: I'll ask. Was there full-scale critical heat flux?

MR. HOSHI: Yes.

MEMBER MAYNARD: You need to come to a microphone and identify yourself, please.

MR. HOSHI: My name is Masaya Hoshi, MHI.

We have conducted a generic test three years ago in the Columbia University. It's a full-scale test.

MEMBER BANERJEE: Full-scale?

MR. HOSHI: Yes. Full-scale means five by five grid spacing using the length of heat and length of throughput.

MEMBER BANERJEE: Not 17 by 17?

MR. HOSHI: Not 17 by 17.

MEMBER BANERJEE: Five by five?

MR. HOSHI: Five by five. I believe that those are the testing authorities used in this industry

MEMBER ARMIJO: Now, just an observation. You are going to 97 percent theoretical density on pellets and probably the highest gadolinia. Later when we get into the details, we want to understand how that is affecting the stress on the cladding, even though it's operating at pretty standard powers.

MR. PAULSON: Okay. We will keep a note of that.

MEMBER SIEBER: It takes a lot of detail to answer that question.

MEMBER ARMIJO: Yes. That's why I just said we would do it later.

MR. PAULSON: Just to look at the standard designs, I think. I don't think that there is anything particularly new here. So I am not going to go through it. But it's to give you a little information on the specific design of the fuel assemblies. We'll just move on because we are running short on time here.

This slide, it's fortunately simple and I think critical in terms of how you look at safety in the place because I think the key parameter here if you want to consider a key parameter is the change in the kilowatts per foot in the design from the standard design, the four-loop that we had, where we improved it in the APWR and improve it once again even in the U.S. APWR. And that's one of the reasons for the improvement in safety in this plant design.

Core design is a low-power density core with flexible operation. We're planning on this core going to 24 months. It uses two batch cycles. Uranium enrichment stays below five percent. Burnups we believe can go to 62 gigawatts, gigawatt-days per ton. Thermal design margins are high in this plant. And, therefore, the peaking factors can go up higher.

Negative reactivity feedback is pretty standard, where their negative feedback for Doppler and moderator coefficient, temperature coefficient, is also negative.

And the neutron steel reflector is added to the design an has two major benefits. First of all, it reduces slightly the enrichment requirements. And, secondly, it reduces the fluence on the vessel.

So it has two significant benefits, very simple design. You will see it a little later on. But it also has both a safety and an economic benefit. That's kind of unusual in safety space.

I mentioned the neutron reflector. This is a series of, it is a build-up series of, pieces that are stacked on one another. There is a significant reduction in the number of bolts associated with the design. If you look at the number of bolts in the baffle/former design, it's like 2,000 that says maybe 50 bolts in.

So in terms of the number of components, it's significantly different. It's very simple. There's no magic about this. It's just basically a block with holes in it. And so the thermal hydraulic characterization, characteristics on this are very simple, but it has we think a significant benefit to the overall design, both in terms of economics and safety.

You have asked from a computer code point of view what we are doing different than what you have seen in the past. What you are going to hear is an answer to that. There are two codes that are specifically Mitsubishi codes, both in fuel. Jeff mentioned the FINDS code as one of them. We also have the FINE code, which provides some of the fuel parameters and characteristics.

We have provided topical reports for these. And so you will have a lot of information to review.

MEMBER ARMIJO: And, in particular, those codes you are applying for a generic approval by the --

MEMBER SIEBER: Yes.

MR. PAULSON: Yes.

MEMBER ARMIJO: Okay. We will definitely want to look at those.

MR. PAULSON: In terms of the methodologies that are used for a nuclear design, we used PARAGON, which is a 2-D lattice physics code. I think you have seen that already. Thermal hydraulics, we're using VIPRE and WRB-2 correlations. And those are familiar also and --

MEMBER BANERJEE: Those already you're

using approved codes or --

MR. PAULSON: Yes.

MEMBER BANERJEE: -- VIPRE? Okay. So that's a Westinghouse code, right?

MR. PAULSON: Right. And the RTDP code also thermal design procedures, statistical evaluation I think has been something you have reviewed also.

MEMBER SIEBER: Right.

MEMBER BANERJEE: You are not using NOTRUMP and things like that, just VIPRE?

MR. PAULSON: No. Just VIPRE.

MEMBER BANERJEE: Okay.

MR. PAULSON: The reactor coolant system. Going through the key parts of the reactor coolant system, they are slightly different. One is the larger diameter and larger height of the reactor vessel from a standard four-loop plant. It's the same, by the way, as the APWR. So it's not something that has not been evaluated for many years already in Japan.

MEMBER SIEBER: Does that have the same number of welds as the current plants?

MR. PAULSON: Roughly, yes. It doesn't have any -- well, it doesn't have a weld in the beltline. So it's --

MEMBER SIEBER: Thank you.

MR. PAULSON: I haven't counted them, but I know that there is no significant difference.

MEMBER SIEBER: Okay.

MR. PAULSON: Yamauchi-san, do any of the Mitsubishi know? I think it's about the same number of weld as a standard four-loop design, but I don't want to say that.

MR. YAMAUCHI: I think so.

MR. PAULSON: Yes. Okay.

MEMBER ABDEL-KHALIK: The power-to-flow ratio is higher than the current four-loop plant. How does that affect the propensity for axial offset?

MR. PAULSON: Would one of the fuel people address that? Oshi-san? There was a question on the power-to-flow ratio being higher. And how does that influence the axial offset? Do you know?

MR. HOSHI: My name is Masaya Hoshi, MHI.

The U.S. APWR flow rate is part of the flow ratio. I mean, the flow ratio is a little bit higher than the other standard plants. It grows to some margin to some margin. And that's the only thing that we can think of on the -- there are almost no differences between those two.

MR. PAULSON: No difference in axial

offset?

MR. HOSHI: No differences.

MEMBER ABDEL-KHALIK: The ratio between power and flow is higher in the U.S. APWR than it is for the current loop, for the current four-loop. Even though the flow is higher, the power is higher --

MR. PAULSON: The ratio is higher by --

MEMBER ABDEL-KHALIK: Yes.

MR. PAULSON: He's saying it's close, but he doesn't think there is an impact on --

MEMBER MAYNARD: I think we need to be careful about drawing conclusions right now. And we will have a chance to talk about that in more detail. You can give some thoughts now, but I don't take that as an official --

MR. PAULSON: Right. We reserve the right to change our mind.

MEMBER MAYNARD: Right. And I think it's something that we're going to be interested in when we get into the fuel design and the fuel operation.

MR. PAULSON: Right. Any questions we can handle in a more detailed meeting like that. That's a fairly detailed question. But just off the top of our heads, we can't think of a reason why it should. But it shouldn't be major in any case.

Let's see. I think that's about it. I did mention that there is a difference in this difference in flow. This is the model 100 pump. This is the model 93A pump. And pressurizer volume, there is an additional margin in the pressurizer, which provides a little additional comfort level with respect to events that look at over-pressurization. It's nice to have a little additional margin. And that is built into the pressurizer volume.

MEMBER SIEBER: And trips, too. You don't want to lose the level.

MR. PAULSON: Yes. Good point. This is a little complicated, but I think there are just a few points here. As we mentioned, this is a 4 by 50 percent. For large-break LOCA, this is the high-head pumps. The high-head pumps have direct vessel injection.

Advanced accumulator injects into the cold leg. So if you can follow through on that, it's a little hard in here to see the safety injection pumps pumping here into the vessel, but you can see on the green lines and coming in directly to the vessel. And you can see the advanced accumulators, which are the red dots here going directly into the cold leg.

You can see also that all of the pumps are outside containment but take suction from the refueling water storage pit inside containment. So that's the suction coming in each of the lines. You can see that they're --

MEMBER BANERJEE: So the high-head pumps are going into the hot leg or what?

MR. PAULSON: No, no. The high-head pumps go into the vessel.

MEMBER SIEBER: Right.

MEMBER BANERJEE: Into the vessel?

MR. PAULSON: Direct vessel injection, yes.

PARTICIPANT: Yes. After the injector, we have change, right? That is under two after the separation.

MEMBER BANERJEE: So it automatically switches, right?

PARTICIPANT: No. Manual.

MEMBER BANERJEE: Manual switches. Okay. What time is that?

PARTICIPANT: About four hours later.

MEMBER BANERJEE: Okay. Makes sense.

MR. PAULSON: This is your first introduction to see what the advanced accumulator unless you've peeked at some of our previous literature, what it looks like with respect to performance.

The advanced accumulator is identified as advanced only because it uses a system where flow can be changed in the design. So there's a time period at which there is a high flow once the injection starts and which continues to a point here, at which time it switches to lower flow. And this is accomplished by using a standpipe, which I will show you in some additional slides what this specifically looks like.

But it is an early injection of a lot of water into the vessel. It is intended to fill the lower plenum and the downcomer so that immediately with as high flow level from the advanced accumulator performs that function and then later on performs a function of supplying additional flow to maintain the water level above what is necessary for the LOCA evaluations. And then at some point in time, you can see that most of the; in fact, all of the, flow comes from the high-head pumps.

MEMBER STETKAR: What is the accumulator injection setpoint, the accumulator injection pressure?

MR. PAULSON: Well, the pressure is proprietary information, but it's a nitrogen blanket.

I can say that. But it is in our documentation.

MEMBER STETKAR: Okay. Thanks.

MR. PAULSON: So, in any case, you can see that what the accumulator does, this is how the U.S. APWR performs with respect to what would have occurred in the old design because this has both the low-head pump and the high-head pumps. Here you have no low-head pumps but only the safety injection, high-head safety injection, and the accumulator.

The question is, why did we go to this arrangement? We went to this arrangement because we wanted to use -- and you will see this later on. We wanted to use turbines, rather than diesel generators, a little longer start time but highly efficient for the plant. And it provides us the margin that we needed early on to get additional water into the system for LOCA evaluations.

MEMBER CORRADINI: Instead of making a bigger accumulator? It is a bigger accumulator.

MR. PAULSON: Well, this is simpler because you can control the flow that you need when you need it.

These are the features of the advanced accumulator. And, by the way, there has been a fair amount of scale testing that has gone on for this, went on back in the mid 1990s because this was going to apply and is applied as a part of the APWR design in Japan.

But, in any case, the major components are the anti-vortex cap, which if you had a chance to see some of the testing that went on, which was visual testing, you could see how this performs with respect to maintaining the flow as the flow shifts.

The flow shifts, as you can see in this chart. The flow shifts once the level of the accumulator goes below the standpipe. Once it goes below the standpipe, I will show you in a little later slide, but you get only flow in one direction. When you have flow through the standpipe, you actually get it in two directions because you are seeing flow coming in through the small flow pipe and from the standpipe.

Once the level of the fluid in the accumulator goes below the level of the standpipe, you only get flow through the small flow pipe. That is the one that actually uses the vortex chamber that provides flow to the primary system but regulates that flow using the vortex.

An example of what I just said, water level is reduced. Water level is injected, both two different ways. It comes in. You can see it here. This is a side entrance, which provides an angular momentum to the fluid coming in through the small inlet. This is coming in through the main pipe.

The sum of these two, then, becomes the initial water that goes into the vessel. What this does, basically by having two flows this way, there is no angular momentum assigned to the water once it gets into the pipe. So it flows directly into the primary system.

Once you get to the point where the level drops below the standpipe, you get flow only coming in the side inlet. And it utilizes the vortex to provide flow into the primary system.

MEMBER ABDEL-KHALIK: So gas can actually enter the primary system before the accumulator is fully discharged?

MR. PAULSON: No. Well, we've looked at that. Part of the testing we have looked at is if gas can injury. We haven't found any, but we did look to see if we used saturated fluid, saturated with nitrogen, to see if there was any impact.

And the impact was very small, but the arrangement is such that there is no significant amount of gas that can get into the primary system.

And there is a topical report that discusses it specifically.

By the way, this was high interest by the NRC also. So we turned in our advanced accumulator topical as one of the earliest topicals. It went in about the time of our QA.

MEMBER BANERJEE: So the level can never fall below the side inlet, right?

MR. PAULSON: That is right.

MEMBER BANERJEE: How do you ensure it never falls below the side inlet?

MR. PAULSON: Well, the available flow that you have is such that it continues to -- well, when you say, "never falls below the side inlet," eventually the side inlet goes to zero, but there is always a head of water in the pipe which prevents gas from going into the primary system.

MEMBER BANERJEE: So how do you ensure that head of water?

MR. PAULSON: With the arrangement of the pipe that is connected to the primary system.

MEMBER ARMIJO: Is there only one or are there several of these accumulators?

MR. PAULSON: Four.

MEMBER ARMIJO: Four?

MR. PAULSON: There's one for every loop.

MEMBER BANERJEE: So let's say that the nitrogen pushes out all the water and the side inlet is uncovered. Now, is there no way that nitrogen can then bubble up into that line?

MR. PAULSON: I'll let the expert handle that.

MR. SHIRAISHI: My name is Tadashi Shiraishi. And I am the inventor of the advanced accumulator.

Well, you know, the standpipe prevents the nitrogen gas entrance into the injection pipe. So there is no gas entrance if that is my --

MEMBER CORRADINI: Can I ask the question differently? And then we can stop.

(Laughter.)

MEMBER CORRADINI: If I have a continual loss of pressure in the primary system, I can't understand how eventually I'm not going to have gas flow-through.

I mean, if you equalize it, then I understand. But if I have continual loss of pressure in the vessel, eventually that nitrogen is going to make it in. Okay? All right. Then we're on the same page.

MEMBER MAYNARD: I think it's obvious to say that this will be an item of interest to the ACRS review, also the topical report. And I think we will have a chance to review some things in more detail.

Let's go ahead. Let's go ahead and move on, yes.

MR. PAULSON: Okay. We can move on from there. So in terms of the design features of the high-head injection system, it's four independent trains. And sufficient capacity for safety injection meets safety injection requirement for the core reflooding stage.

The difference between it I think we've talked about already. It's the difference between two trains and four trains, 2 100 versus 4 50 percent. The design is such in all of these that we believe that we will be able to operate with a single failure and one pump out of service. And that's part of the evaluation that we've put into our DCD. And it's being evaluated by the NRC.

There's one additional feature of the plant. And it's one of those additional things that we don't utilize as part of the chapter 15 analysis, but it's a feed-and-bleed capability that utilizes the safety injection pump as a way of providing long-term cooling, if necessary, without using other systems.

This just shows the arrangement of the in-containment refueling water storage pit. As I mentioned, the refueling water storage pit for both the RHR pumps and --

MEMBER ABDEL-KHALIK: Can we go back to the previous slide, please? You used natural boron. Is it natural boron or enriched boron?

MS. ISHIDA: Natural boron.

MEMBER ABDEL-KHALIK: Okay. So what is the critical boron concentration for a fresh core?

MS. ISHIDA: Mutsumi Ishida from MHI.

The boron concentration at the fresh core, at the first cycle, is the most highest one. And it is more or less 1,000 ppm.

MEMBER ABDEL-KHALIK: A thousand ppm?

MS. ISHIDA: Yes. It is because of the use of -- it is because we use a lot of bundle observer.

MEMBER ABDEL-KHALIK: Okay. Thank you.

MR. PAULSON: Okay? All right. Move on. As I mentioned, the refueling water storage pit, both the RHR system, which also is the core spray system as part of the LOCA analysis, and the safety injection pumps take suction from the refueling water storage pit. These pumps are all outside the containment but inside the reactor building, which is the protected building for safety systems.

One of the important parts of this design is that it's segmented into four designs. So most of the systems, electrical and most of the mechanical systems, are partitioned in such a way so that they are in one quadrant of the reactor building.

We have conservative countermeasures. We think for 191, it was interesting that we're coming on the same day as you are dealing with this. We have had numerous discussions with the NRC on this subject.

We think we have a very robust design for the system. We have four redundant passive strainers, sufficient surface area available for that strainer design. We're going to use very low-debris type material. We're not going to use -- we're going to minimize the amount of fibrous insulation and utilize primarily metal insulation and also to avoid problematic chemicals. We're doing some chemical testing this fall. NRC is planning on coming to view those tests to confirm that the strainer design is adequate.

MEMBER ARMIJO: What is that buffer, sodium TB, on your chart? I'm just trying --

MR. PAULSON: Yes, I know. I know. Sodium TB?

MEMBER CORRADINI: You have to identify yourself and say it louder. I'm sorry.

MR. HAMAMOTO: This is Hiroshi Hamamoto from Mitsubishi.

MEMBER CORRADINI: And louder. I don't think the recorder got it. Repeat it, please.

MR. PAULSON: Repeat your name.

MR. HAMAMOTO: Hiroshi Hamamoto.

MEMBER CORRADINI: And the answer?

MR. HAMAMOTO: Sodium hydrate tetraborate.

MR. PAULSON: Emergency feedwater system has four pumps. There are two turbine-driven, two motor-driven pumps. They're each dedicated to one steam generator unless one of them is out of service.

Then there are cross-links for those. There are two separate pools from which the water can be drawn, each 50 percent pools. They also are connected so that any of the pumps can get water from either of the sumps.

MEMBER SIEBER: In four separate fire areas.

MR. PAULSON: Four separate fire areas. Well, they are separated, but they are not separated into quadrants. You can look at that when you see the reactor building.

MEMBER SIEBER: If one catches fire, will it set another one on fire?

MR. PAULSON: No. I think they are fire barriers.

MEMBER SIEBER: It's walls, as opposed to --

MR. PAULSON: Correct. Quadrants, right. The emergency feedwater system is a 4-train configuration, as I mentioned, 2 motor-driven, 2 turbine-driven, each 50 percent. Two safety-grade independent feedwater sources are available. Those two pits are both 50 percent pits, as I mentioned. So I think I have covered most of the material on this slide already.

MEMBER STETKAR: Does the water capacity in the pits combined have enough for 24-hour decay heat removal?

MR. PAULSON: Yes. It's designed for 2,400, I think, at out standby.

MEMBER STETKAR: Thank you.

MR. PAULSON: I mentioned the gas turbine earlier. This is always an interesting subject because it's somewhat new for the United States. The approach is a little bit differently normally.

We have gone to this for a number of reasons. We think that these turbines are highly reliable. They haven't been used for nuclear applications, but they have been used for other applications.

And we have gone through a series of tests that we're providing a technical report on that will be used by the NRC to evaluate this type of pumping system.

But we use it because of the reliability that we have and the ease of maintenance. They are very simple pumps compared to normal pumps. So there are advantages to this. We know it's a subject that may be of interest to you also, and it is being reviewed closely by the NRC also.

We have also, by the way, had a training seminar for the NRC on these pumps that we completed a month or so ago.

MEMBER BLEY: They are turbine generators?

MEMBER STETKAR: Turbine generators, right?

MR. PAULSON: Turbine generators. I'm sorry.

MEMBER STETKAR: I understand.

MEMBER BLEY: Does that information have extensive testing? I'm familiar with some other previously used emergency gas turbine generators whose reliability to start wasn't so great.

MR. PAULSON: We're aware of that data, actually. And I think we have data that supports that this is high-performance --

MEMBER BLEY: Docket.

MR. PAULSON: Right.

MR. KAWANAGO: This is Shinji Kawanago from MHI.

We have already submitted one technical record, which included our reliability data and especially other emergency gas turbine system.

MEMBER ARMIJO: Have you used such turbine generators in your Japanese plants?

MR. KAWANAGO: I am again Shinji Kawanago.

In Japan, we have one experience, only the one experience, to supply the gas turbine engineering to the other emergency system for the nuclear background.

MEMBER ARMIJO: It has operated with license in Japan?

MR. KAWANAGO: Yes, but it is not a commercial nuclear power plant. It is a test nuclear power plant.

MEMBER SIEBER: We will all ponder reliability where applied.

VICE CHAIRMAN BONACA: How many gas turbines do you have? Four?

MR. KAWANAGO: Four, yes. Other emergency --

VICE CHAIRMAN BONACA: Or emergency.

MR. KAWANAGO: In addition, two of the --

MEMBER MAYNARD: Where is the fuel supply for this kept? Is it big tanks or --

MR. PAULSON: Tanks.

CHAIRMAN SHACK: This is seismically qualified?

MR. PAULSON: It has to be. It will be in part of the building that is considered an extension of the reactor building.

MEMBER MAYNARD: By gas, I take it you're talking either natural gas or propane?

MR. KAWANAGO: In a few areas, we use the same fuel, diesel generator. That means that --

MEMBER MAYNARD: Okay.

MR. KAWANAGO: We use this gas turbine, but a few areas of --

MEMBER MAYNARD: Like an airplane engine.

MEMBER SIEBER: It gets into the combustion chamber.

MR. PAULSON: Let's hope.

MR. KAWANAGO: Kerosene or diesel.

MEMBER MAYNARD: Right, yes.

MR. PAULSON: Just some of the benefits of using the gas turbine. I think I have identified most of these already, but there is a space benefit, no cooling required, it's easily maintained.

It's a very simple system. We don't think that that is true of the diesel generator and the performance and reliability and start time now. The start time is the issue we're dealing with specifically with the advanced accumulator. Okay?

Forty seconds is typical, but in the analysis that we perform for LOCA, we use 100 seconds so that there is significant margin between the 40 seconds we think is the right number for this start time, as opposed to what we used in the safety analysis.

There is not too much significant I think about this. You have probably seen that slide a number of times already with respect to the design of the containment. This is a pre-stress concrete vessel. And the design is very standard, I think. It does have a liner.

And it's a fairly large containment. I think it operates under a slightly negative pressure, if I'm not mistaken.

MEMBER SIEBER: Yes.

MR. PAULSON: And the size of the containment is indicative of the size of the components and so forth that we're using for the plant. Just, by the way, the design pressure for the containment, since you'll see it later on in any case, is 68 psi, psig.

The methodologies. You asked about computer codes and are interested in them. I think that this has been addressed already by the NRC in their presentation, but just to mention the fact that these are not computer codes that you have -- you have seen these computer codes already I think is the best way to say it.

WCOBRA/TRAC I think and ASTRUM are known to the NRC and approved by the NRC for large-break, for small-break MRELAP. Now, M means that Mitsubishi has made slight changes in the code to account for, for example, the advanced accumulator and for direct vessel injection.

Those are the only changes that we have

made to it from what you have already reviewed. So that the only significant differences from what you would see would be for the modeling of those systems that are slightly different from what has been used past in the those codes.

Mass energy release, the same comment applies for SATAN or WREFLOOD and GOTHIC. The only modifications are relative to the design changes that we have used. Containment pressure, GOTHIC once again is a widely known code and I think one that has been well-reviewed already by the NRC.

For the non-LOCA codes, we are using MARVEL, which is a code a lot like LOFTRAN, actually. It has many similarities. The modifications that have been made to that code from the versions that have been reviewed by the NRC, which goes back, by the way, quite a ways, goes back into the mid 1970s, in any case, the modifications have been to take it from a single loop or two-loop configuration to a four-loop configuration as somewhat similar to what LOFTRAN has done going from one loop to four loops. Other than that, it's basically the only change. TWINKLE I think has been reviewed on numerous occasions and VIPRE also. The sump channel analysis are all codes that have been reviewed by the NRC. RADTRAD, PWR-GALE I think are widely used in the industry also for dose evaluation.

Severe accidents. The U.S. APWR achieves a high level of safety comprehensively addressing severe accidents and mitigate the consequences, demonstrate compliance with the NRC regulations, including TMI requirements. We also can demonstrate resolution with respect to unresolved safety issues and high-priority generic items also.

A little complicated design, but it shows some of the features associated with protection for severe accidents. Specifically there is a reactor cavity area under the reactor vessel that can be flooded if there are events that are of concern with respect to penetration of the vessel. So that provides for. There are redundant sources of water for that, for that area.

And evaluations are performed -- I think it's part of chapter 19 -- with respect to performance of this under a number of circumstances, which are listed here: hydrogen generator, core debris, steam explosion, high-pressure melt ejection because of the no penetrations in the bottom and steam generator tub ruptures as temperature-induced tube rupture and molten core-concrete interactions, long-term containment over-pressure. All of these have been addressed as part of our evaluation.

MEMBER CORRADINI: So may I ask just one clarifying question? So the way you have the colors and the cartoon, is it the accident management plan to flood up to the vessel for in-vessel retention or is that just the way it's cartoon-colored?

MR. PAULSON: No. It would be flooded up to that level. This is the level that it would be flooded up to.

MEMBER CORRADINI: So for the purposes of trying to keep the core inside the vessel or just because that is the way the geometry --

MR. PAULSON: It depends on the accident. There are some accidents that you could probably identify that you could penetrate the vessel first possibly.

MEMBER CORRADINI: Okay. So that's more a matter of geometry than a matter of plan?

MR. PAULSON: No.

MEMBER CORRADINI: I'll stop asking.

MR. PAULSON: In any case, we have looked at both cases, where there is a debris that hits the floor before there is any water in there. So the water is not in there to start out with and then floods.

MEMBER CORRADINI: So I will ask my question in general and just leave it there.

MEMBER POWERS: I thought you said you were going to stop.

MEMBER CORRADINI: I didn't say when.

(Laughter.)

MEMBER CORRADINI: The reason I ask that is because in other design certifications we have seen, certain applicants intentionally flood above for in-vessel retention and others intentionally keep it as dry as possible. So I'm trying to ask you, are you either or potentially both? So what is your intent from a design for accident management standpoint?

MR. PAULSON: Kawai-san?

MR. KAWAI: This is Katsunori Kawai of MNES.

We think in-vessel retention has much uncertainty. So we don't expect in-vessel retention to merit.

MEMBER CORRADINI: Okay. And your water management? If I might just ask you to finish your thought process? So your water management depends on the accident sequence or do you want to keep the cavity region dry? What is the thinking process?

MR. KAWAI: Recirculating storage water coming into safety injection pump and spray pump. At first this water goes into the cavity. In case of failure of this water input, we use fire water. This is manual operation. Fire water will be inputted going to the cavity.

MEMBER CORRADINI: Okay. Thank you.

MEMBER BANERJEE: Just a question. It's not a severe accident question. Where are you putting your sump screens for long-term recirculation?

MR. PAULSON: They are right here. Do you see this? That is one of them right there, would be right there. There are sumps so that the --

MEMBER BANERJEE: They are actually the screens within the sump?

MR. PAULSON: Over the sump.

MEMBER BANERJEE: Over the sumps?

MR. PAULSON: Right.

MEMBER BANERJEE: All around? How large are the --

MR. PAULSON: Just over the sumps. There are four sumps that the pumps take suction on. It's overload.

MEMBER MAYNARD: Are they fairly large sumps? That's okay. That's some detail we can get into --

MR. PAULSON: I don't know. I can't define. I don't know the exact volume.

MEMBER BANERJEE: Just one more question. Your insulation is RMI, I hope, and not caliceal or Nukon?

MR. PAULSON: Right. Well, we are going as much as possible to metal insulation, as opposed to fibrous. You would only use fibrous if there were some unique reason for it.

MEMBER MAYNARD: Sanjoy, you were out. There's a specific slide where he covered that. And he cut what materials they're trying to avoid and what materials they're trying to use there.

MEMBER BANERJEE: And you have a buffer.

MR. PAULSON: Yes. You thought the last slide was complicated.

(Laughter.)

MEMBER MAYNARD: Digital I&C folks need to wake up now.

MR. PAULSON: Right. That's right. There it is. It's all right there. I'm not the expert on I&C. I'll give you just a quick review. And if you are interested in specifics, we have an expert here.

There are three separate areas to look at.

One is DAS, which is the Diverse Actuation System, which is part of the protection for common mode. We have the protection area, which is this area. And here is the controlled area.

You can see that there are four separate systems with direct lines up through each of the protection cabinets and then into the control room, which is identified here. And there are also direct lines of information coming down to the remote shutdown panel, which is identified here.

There are, like I said, three basic systems we're looking at: DAS; the protection system; and the control system, which is this area over here.

MEMBER BLEY: A question, but it's kind of related to what Jeff showed us earlier. From what you have told us, this I&C system is now installed in the new plant. And it's funding the same one here.

Jeff's package talked about the topical report on the I&C design process, which is what we usually see for plants. Is the actual I&C system submitted as part of the design cert?

MR. PAULSON: Yes.

MEMBER BLEY: Okay. So the actual system is part of that?

MR. PAULSON: Right. And, in fact, one of the things if you get interested enough, we actually have a simulator of the control room now in the U.S.

MEMBER SIEBER: In Pittsburgh.

MR. PAULSON: In Pittsburgh that was looked at by -- well, NRC looked at it this week. Yes.

MEMBER BLEY: And it's this one?

MR. PAULSON: Yes.

MEMBER BLEY: Oh, cool.

(Laughter.)

MR. PAULSON: Larry, would you like to --

MR. BURKHART: We went to visit this week.

MEMBER BLEY: This week?

MR. BURKHART: Yes, just a couple of days ago. It's not exactly 100 percent the same. And I'll let that up to MHI to discuss. It's we were told very similar. It's the simulator for the plant that has the system now. But it's not a four-loop redundant plant, the simulator. So there are some differences, although we've been told it's very, very similar. So it's not exactly U.S. APWR.

MEMBER MAYNARD: It would be interesting. I think it would add some value.

MR. PAULSON: Pardon?

MEMBER MAYNARD: I was just saying it is

something I think at least part of the Committee would be interested in. And perhaps at some point we would like to go take a look at it, too. But we will talk about that later. That's --

MEMBER SIEBER: You'll like Cranberry.

(Laughter.)

MR. BURKHART: I would just second that. It is well worth the visit to see it.

MEMBER BLEY: One last related question. Is this I&C system analyzed in your PRA?

MR. PAULSON: This I&C system?

MEMBER BLEY: Yes.

MR. PAULSON: As far as I know.

Takashima?

MR. TAKASHIMA: My name is Makoto Takashima.

This I&C system is considered in the PRA, including some kind of factors, including. Our PRA is best on these systems.

MEMBER BLEY: And the topical data report you told us about includes the data --

MR. TAKASHIMA: Yes, topical data is included in PRA report.

MEMBER BLEY: Thank you.

MR. PAULSON: This is the control room

that you would see in the mock-up of the control room and the simulator that you would see up in Cranberry. I think we've talked about it enough, but it shows the different locations and supervisor panel, the operator panel, and the large display panels and the location of where the diverse panel will go in the future.

Just a couple of the features. Microprocessor-based digital technology for I&C, no mechanical relays, complete FORTRAN redundancy, distributed architecture, fully multiplexed and duplicated signal transmission for I&C equipment rooms and main control room, and between the I&C systems, common digital platform.

We actually have a submittal to the NRC on the platform, which is called MELTAC, that you can look at. It's a topical report that has been submitted and a fully computerized main control room, touch screen, by the way. The design is touch screen.

MEMBER SIEBER: Thank you.

MEMBER BLEY: Is there anywhere in operating plants where the kind of displays and touch screen facilities you are talking about are currently in use?

MEMBER SIEBER: Yes.

MR. PAULSON: Do you want to answer that, Takashima-san?

MR. TAKASHIMA: No. We operate this all the same numerus in new plants. No. We have no operating plants.

MR. KAWANAGO: Now we have a plant.

MR. TAKASHIMA: Actual plant we will present later, operating plants.

MR. KAWANAGO: He wants to show on our next slide.

MEMBER BLEY: Okay.

(Laughter.)

MR. KAWANAGO: Thank you.

MR. PAULSON: Okay. The non-safety applications are the following. Do you want to get into this, Takashima-san, as to the history? You mentioned that it came on the next slide, and there were certain things maybe you could mention.

MR. TAKASHIMA: I'm sorry. It's on the next slide.

MR. PAULSON: This shows, as I mentioned early on, we were going to show the development history to show that we weren't just shoving this into the U.S. APWR, that there has been a history over the course of roughly the last 20 years of developing the design and also implementing it, primarily in the non-safety systems.

But now later on we're moving very, very swiftly into the safety systems. And the first installation of the design has been completed. The installation has been completed in the Tomari design.

I mentioned the MELTAC platform. This is developed by a sister part of Mitsubishi called MELCO, Mitsubishi Electric Company, and is currently under review by the NRC. The application in Japan has been for emergency safeguard features that will be at Tomari. Another application will be at Tsuruga. This is the APWR design.

Ikata 1 and 2 in 2009, Takahama units over a period of time, and Ohi also, where there will be upgrades to the package well prior to the use of this in the first U.S. APWR.

MR. BURKHART: And, Keith, just to make a clarification on the simulator in Warrendale, we were told that the simulator is the Ohi simulator. Is that correct?

MR. TAKASHIMA: It's based on Ohi 1 and 2.

MR. BURKHART: Okay. Thanks.

MR. TAKASHIMA: Three and 4. Excuse me.

MR. KAWANAGO: And the other question, basically the unit 3 -- my name is Shinji Kawanago.

These are from the bottom three: the Ikata, the Takahama, and the Ohi. This one is actual operation up front. And we have the plant to upgrade by using this same platform.

MR. PAULSON: A significant amount of V&V testing went on for the development in Japan. This outlines some of that. The development began in '97, the V&V testing, three times the Japanese utility ship operators from 12 sites, full-scale simulator.

MEMBER BLEY: Is there a technical or topical report on the operator performance in these tests?

MR. TAKASHIMA: All together describes these kinds of experiences.

MEMBER BLEY: I didn't understand.

MR. TAKASHIMA: Yes.

MR. PAULSON: And we plan on doing basically the same thing. As we said, the first application was in Tomari, Tsuruga, and Ikata for modernization. Let's move on to the next slide. We plan on doing something very similar in the U.S., where we are going to do the human interface, human-systems interface, verification, and validation with U.S. operators.

We will do a dynamic evaluation will be

performed for full-scale for eight U.S. operating crews in the following areas. We will establish a design specification and, with those results, will provide a technical report this year. The NRC staff will visit into the mapping site. We put down that they visited it two days ago and demonstrated plant operation for that simulator.

Well, conclusions are similar. The U.S. APWR is based on the APWR. The reason that is important is because of the amount of testing that went on on the APWR that I mentioned earlier. The U.S. APWR, 1,700 megawatts, which is primarily due to the improved performance of the turbine. It's using known technology that we will do some additional testing on to validate the performance of the turbine.

And the U.S. APWR has been designed to meet all U.S. utility requirements and all U.S. safety requirements, as indicated in reg guide 1.206.

MEMBER MAYNARD: I would like to go back. Now we have a little bit of time if anybody has any questions on what we have gone over, a little more discussion. Sam?

MEMBER ARMIJO: Yes. I may have missed it, but when you had your comparisons of the current U.S. four-loop plants with the APWR and the U.S. APWR, could you tell us the temperatures and pressures of the APWR compared to, let's say, the conventional U.S.? Specifically, have you increased T-hot and pressure in order to get more efficiency?

MR. PAULSON: The pressure is the same, 2,250. Temperatures are comparable, right.

MEMBER ARMIJO: Right.

MR. PAULSON: I'm trying to think what the hot leg temperature is. The inlet temperature is around 555 Fahrenheit.

MEMBER ARMIJO: And T-hot?

MR. PAULSON: I think T-average in the core is 587. So just take that ratio. It's 32 degrees. It puts it right in the range of known operating plants.

MEMBER ARMIJO: Thank you.

MR. PAULSON: Yes?

MEMBER STETKAR: Did you mean T-ave is 587? That is a little high.

MR. PAULSON: That is in the core. That is not exit.

MEMBER STETKAR: You said T-ave. Did you mean T-ave?

MEMBER SIEBER: That is what he meant.

MR. HOSHI: My name is Masaya Hoshi.

I am talking about that the temperature temperature. We are keeping the T-hot as 325 DUC. That is our common design. So APWR and the U.S. APWR, those two plants are the same T-hot designs. The flow rate is different. So the T-operation, T-inlet might be different, but T-hot is the same.

MEMBER MAYNARD: That makes it very comparable to what the current U.S. --

MEMBER SIEBER: Yes, that's what he said.

MEMBER MAYNARD: -- 555, 585 to T-ave and about 617 or so for T-hot.

MR. PAULSON: It's about the same. I think that was the question, was the pressure and the temperature about the same.

MEMBER MAYNARD: Right, right.

MR. PAULSON: And the answer to that is yes.

MEMBER SIEBER: It's within a degree of the upgraded Millstone plant.

MEMBER BANERJEE: I have a question on your slide where you showed ECCS and CSS/RHRS if you can go back to that.

MR. PAULSON: Do you have the number on it?

MEMBER BANERJEE: I can't see number.

MR. PAULSON: It's a little number in the lower right-hand corner, that last number.

MEMBER BANERJEE: Seventeen. If you look at that, it seems -- since there is no scale, it is hard to say, but it seems that you have lower amount of flow in the long term anyway that is your current strategy than when you had a low-head injection pump with the control flow loop design. Is that true or is that just an optical illusion based on not having --

MR. PAULSON: How is the flow compared to the standard four-loop design long term compared to what we have?

MEMBER BANERJEE: Not even long term, even relatively in the short term. Since there is no scale, it is hard to know. Yes. It goes from about, you know, wherever that green thing takes over on the left-hand side.

MR. PAULSON: We can get an answer to that question. I think there was much different. It had to meet cool-down requirements.

MEMBER BANERJEE: Yes. So I don't know what. Since there is no scale, it is impossible to tell, but qualitatively it seems that you are going to have less flow.

MR. PAULSON: Well, some of the scales on this are proprietary, but they're in the report.

MEMBER MAYNARD: I would think it would be lower until long term, when you end up putting RHR on for the longer-term cooling because compared to the current, you've got RHR and your is pumps. And on the U.S. APWR, you only have the safety injection pump. You don't have the RHR at that point. So I would think the flow would be --

MEMBER SIEBER: The available flow is lower, but it meets the requirement.

MEMBER BANERJEE: It may meet the requirement, but, nonetheless, I mean, if we could see quantitative. At some point we are going to see some quantitative numbers --

MEMBER SIEBER: Right.

MEMBER BANERJEE: -- on pressure, injection pressures, all these things.

MEMBER SIEBER: Right. Get that in closed session.

MEMBER BANERJEE: When is that going to be? I mean, hopefully not too far down the line so we are apprised of what is really different about this design because this is very different not having a low-pressure injection system. We're seeing this in another design, where we don't have a high-pressure injection system.

MEMBER MAYNARD: Every one of the revolutionary designs is going to have something that we're not --

MEMBER BANERJEE: Is it safe, though? I always feel that the more water you can get in, the better.

MEMBER SIEBER: Yes, until --

MEMBER MAYNARD: We want everything.

MEMBER SIEBER: It depends on what the cost is.

MEMBER MAYNARD: We will be looking at schedules on what time we look. You know, some of this, as far as the chapter review stuff, that is going to be quite a ways down the road.

Some of the topical reports, some of the specific topics, we're going to be looking at what do we need to be looking at sooner with various subcommittees and things there.

MEMBER ABDEL-KHALIK: What codes do you use to analyze subcooled boiling, crud deposition, and boron deposition in the hot channels?

MR. PAULSON: Who would be the best to answer that?

MR. KIKOTA: Excuse me? Could you --

MEMBER ABDEL-KHALIK: What codes do you use to analyze subcooled boiling, crud deposition, and boron deposition in the hot channels?

MR. PAULSON: Computer codes.

MR. KIKOTA: My name is Michitaka Kikota, MHI.

Boron, we use an input calculation code which is made for following the NRC requirement. What is the question? I cannot understand.

MEMBER ABDEL-KHALIK: I am trying to figure out, how do you determine how much boron actually or how much crud deposits in the upper part of the core and how much axial offset do you get?

MR. PAULSON: How do we calculate crud deposition and how much is it? Do we know? Do we have a --

MR. TESHIMA: I am Hideyouki Teshima from MHI.

With regard to axial offsets, I think there are three factors for the AOA. The first one is the solution of the crud.

MEMBER ABDEL-KHALIK: I understand. I am asking what codes do you analyze that.

MR. TESHIMA: In the FIND codes, the raw design code, we assume some of the crud deposition in the calculation.

MR. PAULSON: That is one of the unique codes. And it's one that we have written a topical on. That is available.

MEMBER APOSTOLAKIS: Can we go to slide 40? Could you explain a little bit what you mean by HSI verification and validation has been conducted and then the second bullet as well?

MR. PAULSON: Bullet 1 and bullet 2, what we're planning on doing --

MEMBER APOSTOLAKIS: Briefly what --

MR. TESHIMA: We are now planning verification and validation of our standard HSI design and standard human considerations by U.S. operators from Luminant.

MEMBER APOSTOLAKIS: What does it mean to verify and validate?

MR. TESHIMA: Now we plan to do six actions: CTR events, heat-up and cool-down of the operation, these kinds of simulated operations. We operate by U.S. operator and we validate/verification. We provide verification and validation of our design.

MR. KAWANAGO: In addition to verification --

MR. TESHIMA: Oh, verification and

validation of our standard design of the HSI systems.

MR. KAWANAGO: Validation and verification.

MR. TESHIMA: Oh, verification and validation. Okay. I think verification and validation means U.S. operator using our simulator to treat after the accident. Okay? By using the actual simulator and touch our display and using our system on those, using our main control board.

MEMBER APOSTOLAKIS: That sounds more like a training.

MR. TESHIMA: Yes, yes. No, not training.

MEMBER APOSTOLAKIS: How is it verification? I don't understand what verification means.

MR. TESHIMA: On each step operator using our display. And our display, also scheme, total design of our main control board is enough to apply U.S. operating plant by verify by U.S. operator.

MR. KAWANAGO: This is Shinji Kawanago.

I will try to explain a little bit. Now, as we have explained, we already have the actual design of a system. We have already applied to the Japanese nuclear power plant. And so we already have the actual display and also procedure how to use the main control board, feature private computer, computerized system.

However, -- and we know to apply that to the design. And to the U.S. operator and U.S. operator have some preference and also some specific request for us.

So maybe the culture between the Japan and United States, there is some difference. And so we need to verify that our actual design is applicable to the U.S. operator or not. And maybe there is some modification we need.

And so we call that one verification and validation by U.S. operator.

MEMBER APOSTOLAKIS: Now, if we go to the fourth sub-bullet under the second bullet, "Normal and degraded HSI conditions" --

MR. TESHIMA: This meaning the -- we have two types HSI systems. One is non-safety systems, normally operator using non-safety systems. That's non-safety systems can operate non-safety and also safety equipment. But this is a non-safety system.

And if we assume a total failure of the non-safety system, we still have safety, safety-grade HSI. So the safety can keep the safety I&C. We want to demonstrate these kinds of situations.

MEMBER APOSTOLAKIS: And there will be some report of this?

MR. TESHIMA: Yes. And also we provided the actual report of the result of these kinds of operations by the end of this year.

MEMBER APOSTOLAKIS: Thank you.

MEMBER ABDEL-KHALIK: Can I follow up on George's question? What kind of data are you going to collect during this verification and validation process?

MR. TESHIMA: I think the data, first is operator comment. We gather the operator comment for our design. And also we measure the actual operator performance, time, touching time, and the operator performance we measure.

MR. PAULSON: If you look at the information that's supplied on the board, you will see that the procedures are there. So you can see if the procedures are there, you could compare that with what the procedures indicate the operator to do. And you can look at the times.

That is one thing that Takashima is focusing on, is how well can a U.S. operator perform functions associated with the procedures as identified? That is a key measure.

The other issue I think that Mr. Kawanago mentioned is to look specifically at the differences between U.S. performance and what U.S. operators are looking for in terms of information that they need to perform certain functions that are in the procedures?

Those types of things will be recorded also and factored into the final design.

MEMBER ABDEL-KHALIK: So the emergency operating procedures for this plant have already been developed?

MR. PAULSON: There are procedures for operation of the plant that have been developed in Japan.

MR. TESHIMA: Excuse me. On this verification and validation, it would be current, based on current plant. And we also we already have the emergency operating procedures for this plant. So by using these operating procedures, we will have this kind of verification and validation.

This verification and validation process is to check our standard HSI design for operating plant and U.S. APWR. The actual validation and verification for U.S. APWR will be performed later by using actual U.S. APWR simulator and actual U.S. APWR displays and actual U.S. APWR EOP. These kinds of activities will be performed next week. This is based on the standard.

MR. KAWANAGO: This is Shinji Kawanago.

I want to expand a little bit. As we explained to you, we have already had huge experiment data, which was conducted in Japan, to make sure on this human-system interface.

For example, the Japanese operators have already used this one. And we check the monitor, the actual time, during the accident. And so through those processes, we developed this human-system interface.

And so we have already furnished the simulator in the United States. And we want to compare the previous data in Japan and test the data, which is conducted by the American operator. We want to compare.

And if there is no significant difference, that means this human-system interface is good. But if there is some difference, we need to modify our standard design for this U.S. APWR.

After that and, actually, it is we develop the actual design for the U.S. APWR again. But, anyway, of course, we need to do the actual standard design for the --

MEMBER APOSTOLAKIS: It sounds to me like, instead of verification and validation, the appropriate word is "adaptation."

MR. KAWANAGO: Adaptation.

MEMBER APOSTOLAKIS: Verification and validation is --

MEMBER SIEBER: It has another meaning.

MEMBER APOSTOLAKIS: You are adapting it to American --

MEMBER BLEY: Will those U.S. tests be done as part of this design certification or will that come later?

MR. KAWANAGO: Yes. It is part of the DCD.

MR. TESHIMA: This test is part of the DCD.

MEMBER BLEY: Okay.

MR. TESHIMA: So we will provide the report in the DCD.

MR. BURKHART: Well, I recommend when you go to your visit to the simulator that -- we had a very good presentation on what MHI proposes to be submitted and what is going to be done when, although -- correct me if I am wrong, Mr. Kaneda -- I believe the HSI task analysis for the U.S. APWR isn't going to be done until 2012.

So just take away from this I recommend going on that visit because it is well worth it. And MHI can lay out what they expect to submit when. Now, I will tell you that based on our visit, we have some questions.

But I think that would be very revealing on what their intentions are in this area.

MEMBER MAYNARD: One more question, and we will be done.

MEMBER ARMIJO: I presume the fuel for this plant will be manufactured in Japan.

MR. KAWANAGO: Yes.

MEMBER ARMIJO: And I guess the question I have to the staff is, what is the NRC's involvement regarding the fabrication facilities in Japan? Is it hands-off? Is there some sort of an audit or review?

MR. BURKHART: We are already inspecting. We have vendor inspections going on already over in these facilities.

MEMBER ARMIJO: Is it the same thing you would do, for example, from a fuel supply from a U.S. --

MR. BURKHART: I imagine the answer is yes. I don't know if anybody from the QA, NRO's QA

Branch, is here, but we have been actively going out to all of the facilities overseas to conduct vendor inspections. We can provide a separate briefing on that if you'd like.

MEMBER ARMIJO: I just wanted to know what your practice was.

MR. BURKHART: Yes. I can definitely tell you that there's a lot of folks who have been overseas looking at all of these facilities. And we'll continue to look at them.

MR. PAULSON: We have had U.S. audits for replacement components already in Japan. So it's --

MEMBER ARMIJO: From the NRC?

MR. PAULSON: Yes.

MR. BURKHART: We are also looking at --

MEMBER MAYNARD: I see a potential need for ACRS to conduct some visits on our own.

(Laughter.)

MEMBER MAYNARD: I would like to go ahead and bring this to a close. This was an informational briefing. So we'll have opportunity later to go into more detail on a number of these.

I would like to congratulate MHI and also the staff. I think the presentation hit the points that I thought were important to the Committee, identifying the codes and going through the design, very thorough presentation.

I also compliment the ACRS for actually letting them get completely through their presentation and have time at the end for some questions.

(Laughter.)

CHAIR SHACK: It's just because people fear you.

(Laughter.)

MEMBER MAYNARD: I think a couple of obvious things that have come out of this, you know, there are a few of the topical or technical reports that we're going to want to take a look at. I'm sure that the two fuel topicals is something we will want to take a look at here and also the accumulator. I know the staff has had interest in that. That's a topical report I think we're going to want to take a look at.

At our next meeting in July during the planning session, I'm going to try to identify or make sure everybody has a copy of the list. At that time we can talk about what things do we maybe want to take a look at before we start getting the chapter reviews and stuff on that.

Also I think getting a copy of the DCD

will allow us to take a look at -- you know, it may answer a lot of questions that we have. If we see areas where it doesn't really have the detail then we may need to set some different meetings up to take a look at that ahead of time.

MEMBER BANERJEE: And when can we get the DCD and the list of topicals?

MR. PAULSON: Five minutes.

MEMBER BANERJEE: I don't want paper, CD or I prefer it on a memory stick.

MEMBER MAYNARD: Okay. I would like to bring the meeting to a close. Again I would like to thank everyone for an outstanding presentation and an overview. And I turn it back to you, Mr. Chairman.

CHAIRMAN SHACK: Okay. Right on time. I think it's time for a 15-minute break. We will resume at 10:45.

(Whereupon, the foregoing matter went off the record at 10:27 a.m. and went back on the record at 10:46 a.m.)

CHAIRMAN SHACK: I would like to come back into session.

Our next topic is one of our favorite GSIs, 191, pressurized water reactor sump performance. And Sanjoy will be leading us through that.

MEMBER BANERJEE: Okay, I think we all know what the issue is which is that the Generic Letter in 2004, Generic Letter 2004-402, potential impact of debris blockage in emergency recirculation during design basis accidents at pressurized water reactors. And this Generic Letter required PWR licensees to do certain things which I won't go into detail.

Anyway, the end process of all of this is that we have our licensees right now putting in much larger sumps to make sure that we don't get too much pressure drop or sump screen slurry.

Mike Scott is going to update us on these activities and we hope that this matter will eventually be closed out as quickly as possible. ACRS is also getting fatigue on it.

(Laughter.)

Okay, go ahead, Mike.

MR. SCOTT: Okay, thank you. As Dr. Shack referred to as one of your favorite GSIs, it's absolutely my favorite to the point that I won't work on anything else.

(Laughter.)

I am more than pleased to be back to talk to you about this issue. And before I get started, is this supposed to project up here?

CHAIRMAN SHACK: Yes.

MR. SCOTT: What do I do to do that?

MEMBER BANERJEE: Somebody will help you.

(Pause.)

MR. SCOTT: I'll get started. Before I get into the slide show, I'd just like to make a couple of introductory remarks. First of all, we are pleased, I am pleased to present this subject to you again and pleased to provide you some good news.

Substantial progress has been made with regard to GSI 191 since our last talk to the full Committee which was about a year ago. We believe that resolution of the issue -- as Sanjoy referred to, we are closer than we were. We think we're significantly closer than we were a year ago. For example, as was mentioned, effectively all of the PWRs now have significantly larger strainers installed by one to two orders of magnitude. They're larger than what they were when Generic Letter 04-02 was written.

A number of plants have changed their sump buffers typically to sodium tetraborate, although sometimes other buffers, depending on the plant specific conditions which has also reduced vulnerability. Some plants have removed problem insulation. All of these actions have significantly, we believe, enhanced safety in the PWR fleet.

Meanwhile, the staff has, through many interactions, generally accepted most of the vendor test protocols. As you know, we came to you a year ago. There were significant concerns with the test protocols. We have worked through most of that now with some exceptions. There are some second order issues that remain with some of the protocols, but by and large, we believe that the vendors have gotten to the point where they have tests that we believe show conservatively whether a strainer performs adequately or not.

And a number of licensees have reported completion of all corrective actions and they believe that they have satisfactorily addressed Generic Letter 04-02. We are in the process of verifying whether we agree with that.

So that's the good news. On the other side of the ledger, we're still not quite as close to the finish line as we predicted we would be, and as we would like to be at this time. We had an original target date for closing the generic issue in 2007. That didn't happen. And I'm going to talk to you in the presentation about some of the reasons why.

A few significant issues remain, as I mentioned, with a couple of the test protocols. We have to work through that. And even when we work through that, many of the licensees tested with old protocols and those protocols we had issues with and those plants are likely to get requests for additional information from us.

As I believe you all are aware, the downstream in-vessel issue is not fully resolved and I'll talk some about that today.

Some high-fiber plants, we believe, will likely struggle to show success with a test protocol that is clearly conservative. And so it may that at the end of the day some plants need to take additional measures to reduce their vulnerability to this issue. We don't know that at this point, but we have seen some test indications that show that a -- so to speak, a little bit of debris can go a long way.

I believe that Dr. Graham Wallace referred to this issue very appropriately in our last Subcommittee meeting. It's like a hydra. You cut off a head and two more grow back. And that's been a frustrating part of GSI-191. So it's good news, bad news and today I'm here today to talk to you about both and give you a picture of where we're going and when we plan to come back to the Committee.

Slide 2 refers to things that Dr. Banerjee already talked about. Generic Letter 04-02 is our primary regulatory vehicle for seeing correction of the issues posted by Generic Safety Issue 191. And basically, what that Generic Letter said was by the end of 2007, each PWR licensee should determine what its plant-specific debris generation transport are, make needed modifications to show compliance with the regulations and the presence of that plant-specific debris loading and update the licensing bases for the plant to reflect those corrective actions made. That was what was supposed to happen by the end of 2007. We didn't quite get there.

So where are? I talked about the fact that we have much larger strainers and a number of other modifications have been done. You may or may not be aware that the Fort Calhoun plant implemented what we called a water management initiative. That is, they revised their licensing basis such that the containment spray would not be used in the event of a LOCA. It was not needed to be used. And if containment spray is not used, that has several beneficial effects for strainer performance. It cuts significantly on the flow rate of the water that goes into the sump and impinges on the strainer and it also means that a lot less debris is washed down by the containment spray system into the sump. So a plant that chooses to do this can make its burden a lot easier on showing that strainer performance is adequate. And Fort Calhoun, as I understand it, has implemented that change.

We don't know whether other plants plant to do that. We certainly do not have any other submittals from plants, but this was an initiative that the NRC, specifically the Commission encouraged. And at least one plant took.

Staff and industry both believe that the risk of clogging is significantly lower than it was when the Generic Safety Issue was initiated and when the Generic Letter was initiated. And we believe that plants can continue to operate safely for the same reasons that were stated in Generic Letter 04-02 while we work through the remaining issues associated with closing out Generic Safety Issue 191.

Integrated head loss testing including chemical effects is on-going. You may recall that we talked to you a year ago about what the kind of chemical effects testing was going on. We mentioned to you at the time that we had concerns with some of the aspects of the test protocols and as I mentioned to you in my introduction, we have largely worked through those with the vendors and with a couple of exceptions and the staff is observing and commenting on representative tests intended to show function. So we believe we're approaching the end of that process.

Now then the question becomes is those who tested and took credit for earlier testing which didn't pass muster with the NRC, how we will deal with those.

MEMBER MAYNARD: Do we know why more plants haven't used the water management option? Is it more of a regulatory burden or more of a physical design that causes problems?

MR. SCOTT: I honestly don't know. Clearly, Fort Calhoun came in and they got approval for it. It may be that -- and I'm sure this is the case, as with everything else with GSI 191, it's extremely plant-specific. So if your particular -- let's say you're a low-fiber plant. You probably don't much issue here and you don't have motivation to try to pursue a water management change. You might have a particular combination of debris and chemicals such that it's just not worth your trouble to go to. Maybe a particular plant needs containment spray to get through a LOCA. This plant was able to show that they didn't. It may not work for everyone.

Bill wants to comment.

MR. RULAND: As I understand it, Fort Calhoun has a -- for the size of its plant, has a rather large containment and their containment cooler capacity is also a large containment cooling capacity. So coupled those two features coupled together enabled them to go forward with this initiative and the staff did a review of this including the debris generation and how that would affect the containment coolers. But again --

MEMBER MAYNARD: And I understand for some of the plants, plant-specific issues.

MR. RULAND: That's correct.

MEMBER MAYNARD: I would have thought there would have been a few more.

MR. RULAND: And so did we.

MR. SCOTT: And let's think about it in these terms. It is not necessarily the end game yet. A particular plant may have difficulty showing that they've adequately addressed the issue to the staff's satisfaction and they may find themselves in the mode of making additional changes. And this would potentially be on their menu for doing that. But it is accurate to state that so far we only have one.

I indicated that our goal was to be done by the end of 2007. That did not happen because of various issues, downstream effects analyses were and are on-going. The head loss testing was not done, is still not done.

Most of the plant modifications were completed by the end of 2007, but there were a few that remain, particularly with regard to a piece of equipment that would be problematic to get to during a normal operation and which is not likely to cause a problem for a plant.

An example is Diablo Canyon received an extension to January 2009 to remove certain insulation that is difficult to access on their steam generators. They're replacing the steam generators in January 2009 anyhow. This insulation would only be affected by a very few LOCAs and so you look at the risk of it, it's very small and the dose is very large and so it did not seem to be an intelligent thing to do to try to push that to be done before the steam generator replacement. So it's that kind of thing.

One of the plants received additional time to make a number of small modifications to their pumps to reduce their risk posed by downstream effects, blockage of the lines to the pumps or damage to pump seals.

CHAIRMAN SHACK: Let me just come back to that water management initiative again. That presumably is all done in design basis space, right?

MR. SCOTT: Yes.

CHAIRMAN SHACK: Nobody looks at what that does to the PRA.

MR. SCOTT: I don't know the answer to that, how it was processed and whether it was risk-informed or not. I don't.

CHAIRMAN SHACK: We normally encourage plants to have containment sprays for various reasons.

MR. SCOTT: Well, and it's not like they have removed the system. The system is available, but -- and there may be -- and I'll be honest with you here, I'm not familiar with the details of this. They may have some gates that say well under certain circumstances I am going to use it. I just don't have that information in front of me. We can get you, if you would find it helpful, a copy of the license amendment application and the staff's review of it.

CHAIRMAN SHACK: I'd be interested in seeing that.

MEMBER BLEY: Something like SAMGs might

pick this up as a later action.

CHAIRMAN SHACK: Right. I mean you might not need it right now, but some time along the way.

MEMBER BLEY: In some bad scenario you might.

CHAIRMAN SHACK: In some bad scenario.

MEMBER MAYNARD: If you are taking it out of the automatic actuation.

CHAIRMAN SHACK: Right, but I'm not sure that just because I take it out of the automatic actuation that's really solved my problem. When I need to use it --

MEMBER BLEY: Once it's out of automatic operation, is there any requirement that it be there at all? Maybe not any more.

MEMBER APOSTOLAKIS: Even the automatic actuation. You can't just say I will remove the automatic actuation and do it manually. I think it's tied to the diesel, isn't it?

CHAIRMAN SHACK: Yes, but that's all addressed in licensing space.

MEMBER APOSTOLAKIS: But still.

CHAIRMAN SHACK: I'm interested in what it does in a wider range of accident management.

MEMBER BLEY: The PRA says there's a lot

of cases where it's really nice not having it because you conserve water and get a lot longer. Later on, you might really want to have it.

MEMBER BANERJEE: If you will provide this, let's take a look and let's move on.

MR. SCOTT: Okay, right, and we're not prepared to address that subject in detail today.

MR. KLEIN: One clarification, Mike. This is Paul Klein from NRR. They will not take the system out of service because it's still being used for a main steam line break. It's just the auto start on a LOCA, the logic was changed so that you would not have an auto start on a LOCA.

MR. SCOTT: For the record, that was Paul Klein, NRR.

MEMBER APOSTOLAKIS: But these decisions to give them more time are essentially the result of judgment, right? NRC monitored judgment taking into account the facts as you said, the risk is low and they're going to do this anyway? That is basically somebody's job?

MR. SCOTT: That is correct. We actually sent a SECY paper to the Commission in 2006 that specified the criteria the staff planned to use to evaluate extension requests. And it was along the lines of what you're talking about. What are the mitigative measures you have taken. If you want a lengthy extension, tell us about the risk association with it. We believe that a short-term extension of a couple of months or a few months would show up very low in risk base any how. But if you want a lengthy extension, then tell us why it's okay.

There were specific criteria for it, but it's -- you're correct, it is a judgment call on the staff's part. And the other part of the picture was mitigative -- modifications that have been made. Everybody has a larger strainer. They've addressed a lot of the issues. Everyone essentially has addressed most of the issues associated with this now and so we're cleaning up the remaining issues. Typically, an extension would be for one particular mod or to complete the analysis. You know, the way we did this which is very different is do the mods now on the assumption that the old strainers were too small, then follow it up with the analysis that shows that the modification is adequate to address the issue fully with full knowledge going into that that we might find that additional actions were needed.

So now what they're doing, most of them, the mods are done and now they are doing the analyses and these analyses are complex and tricky and sometimes it's difficult to have a demonstrably conservative test that still passes the criteria. So that's what's being struggled with.

MEMBER APOSTOLAKIS: Okay.

MR. SCOTT: We talked about extensions. All plants have now given us a supplemental response to the Generic Letter. We asked for those in February and we received it from all the plants. Some of those responses were incomplete because the plants had received extensions to do additional actions.

Chemical effects. For some time, we considered chemical effects to be the most challenging issue associated with GSI 191. We believe that the test vendors at this point most of them have a handle on how to test for chemical effects and the plants are conducting tests. As noted here, they didn't get done by the end of 2007 for various reasons. The industry, we believe, was a bit slow in recognizing the significance of the issue. The ICET round of testing and the follow-on testing certainly indicated some surprising results that helped motivate action in this area. Once the action was begun, there were only so many test vendors and so the licensees queued up with the test vendors to get their testing done.

As I mentioned to you, the staff posed challenges along the way by raising issues with the prototypicality or conservatism of the test protocols. That slowed them down further as they resolve the staff's issues. And so we ended up not getting done by the end of 2007.

We did issue a safety evaluation on the chemical effects topical report, WCAP-16530, in December 2007. Licensees can choose that report to help them to go through their chemical effects evaluations or as with all these issues that they don't choose to use topical report they can use their own plant-specific method if appropriately justified.

So we believe that the licensees are moving forward on chemical effects and have a path forward to show a successful test of chemical effects issues.

You've heard about the chemical effects peer review. The Office of Research commissioned a peer review in 2006 that identified 100 or so questions regarding chemical effects that they thought had not been answered yet. The staff has gone through a multi-tiered process to screen those peer-review questions to identify those warranting further evaluation and we have and Office of Research has briefed the Committee some time back on the type of questions that were asked and our plans to review them.

We are now reviewing the study results from the staff-supported work intended to screen those issues. The likely result is that there could be a need for additional consideration of some of these effects and the number currently being bandied about is four, four particular effects that might need additional work. However, the staff has not completed its effort in screening those effects. We do expect to finish that work in the next few months and plan to report to the Committee on this along with a number of other subjects later in 2008.

Downstream effects. We divide that into two parts: ex-vessel and in-vessel. We did issue a safety evaluation on an ex-vessel proprietary downstream effects topical report, proprietary ex-vessel downstream topical report. Issued that in December 2007. Some licensees are still working through having to do these type of tests as a result of the fact that that SE came out in late 2007. So some plants had extensions to perform this work.

And then there is the other issue and I mentioned a few minutes ago the chemical effects were thought to be the most challenging issue associated with 191 and 191 always surprises us and now we've had a surprise with regard to in-vessel downstream effects which refers to the potential for blockage of flow in the core and/or local effects in the core.

We did receive a topical report, WCAP-16793 in middle of last year. We issued a draft safety evaluation in March of this year which we provided to the ACRS Thermal Hydraulic Subcommittee. We met with the Subcommittee in March and the Subcommittee had a number of questions and concerns. The staff and the PWR Owners Group are now working to address those concerns and the Owners Group has concluded that it needs to do additional testing.

We are attempting to work with the Owners Group to get them to identify a test protocol that we would view as adequately conservative. There had been on-going discussions about that which is why we have not been able to come back to the Subcommittee and say here is the plan to address your issues. So we do plan to do that, obviously, to get back to the Subcommittee as soon as we have a clear path forward and as soon as we have some information to provide you. We anticipate that will be in the near future. But this -- some issue has sometimes, as I mentioned, surprised us.

So we have, by the way, as a separate data point one licensee who has chosen to test themselves rather than test an association with the Owners Group. They performed a test at CDI facility in New Jersey which the staff observed a week or two back. And the staff found that that test protocol appeared to be adequately conservative. We will be happy to talk to you in some detail about what we observed at that test and hopefully test results from the Owners Group work in the near future. But the lesson we took away from that is that there is a protocol that we believe is defensible. Whether the Owners Group will use it is unknown and the question arises okay, that was with one type of fuel and we have a number of different fuel designs and to what extent does the Owners Group work bound all the fuel that's out there. And we don't know that yet. We have asked them questions about that. We know that the different designs of fuel have a very different geometry at the inlet, all intended to discourage intrusion of debris during normal operation into the fuel. And this is one of those situations where it could actually encourage a debris bed at the inlet. So that all has to be sorted out. It's being sorted out as a high priority now and we will update you all on it when we have more information.

MEMBER ARMIJO: Michael, I have a real quick question. I didn't attend the ACRS Subcommittee on this subject, but in the questions that are being addressed, obviously the temperature and the flow and everything in the assemblies, the fuel assemblies are being considered, but will these chemical change due to radiolysis? Is that being addressed, that question?

MR. SCOTT: Paul Klein can correct me if I'm wrong here, but I believe that radiolysis effects is one of the four peer-review questions that's being addressed. Is that right, Paul, or -- set me straight.

MR. KLEIN: That's correct. One of the questions raised by the Peer Review Committee was the effect of radiation not only on the precipitate, but on metallic corrosion rates. So that is one of the topics of the four that remain.

MR. SCOTT: So those remain on our plate to deal with. Did that answer your question?

MEMBER ARMIJO: Yes, thank you.

MR. RULAND: This is Bill Ruland. However, regarding the in-vessel topical report, the radiolysis is not part of our analysis, is that correct?

The radiolysis is not part of the analysis for the in-vessel topical report. Rather, it's a peer review issue that is subject to further research.

MR. SCOTT: I believe that is accurate, yes.

MEMBER MAYNARD: For the plant that's doing the plant-specific testing, is that just for a very specific fuel -- if they make design changes in the future, they basically have to redo testing?

MR. SCOTT: Well, you could ask that question about any aspect about GSI-191, if you think about it.

I've got like 12 review areas to show adequacy in this issue: coatings and chemicals and downstream and upstream and so on and so on. And there will be a licensing basis for the plant in every one of those areas. And if the plant changes that licensing basis, then we have regulations that, of course, call for them to evaluate that. So if they change fuel types to a different fuel than what has been certified, so to speak for GSI-191, then they're going to have to evaluate that change. That might take evaluation. It might take testing.

MEMBER MAYNARD: What I was really getting at -- I know the approach of the WCAP was to provide a method for evaluating and we're still going to have a plant-specific, design-specific review against that, and I didn't know if this testing provided data for a methodology or whether it would just for what they had in the core right now.

MR. KLEIN: Mike, if I can jump in here. That particular licensee looked at the configuration that they have in the existing core and then they also looked at another configuration that they were considering switching to in the future and they saw that that did make some differences in the pressure drop that was observed.

MR. SCOTT: Which is also information for us when interacting with the Owners Group because we are concerned that if there is more than one type out there, which there clearly is, and if it has a significant impact on head loss, which it clearly does, then how many tests do you have to do to bound all that? That's why this is not trivially easy to do and get right back in to talk to you all. That's kind of where we are.

MR. RULAND: And more generically, you've raised a question that we have also considered, that maintaining the licensing basis for GSI 191 is going to be non-trivial. So the staff is already considering a document, probably a regulatory issue summary that is going to come out that is going to try to provide guidance to the licensees about how to maintain that. Basically tell them, describe how we did the review and provide them guidance on how to maintain the licensing basis.

MEMBER BANERJEE: I have a question about this. I'm aware of the test that you are talking about, but were there any chemical effects there?

MR. SCOTT: The test that we're talking about did include chemical effects.

The last sub-bullet on here refers to the fact that I initially in this presentation had some more discussion on the draft WCAP and some of the staff's conclusions, which actually the Subcommittee has seen before. Because of the time constraints associated with this presentation, I went ahead and moved them back to the backup slides. But that information is there if you are interested in seeing it.

This is a summary of our understanding of the Subcommittee's questions and concerns regarding WCAP 16793. And we did send, I sent Dr. Banerjee and email on that and he actually added a couple of items which are reflected in here. So we believe this is an accurate summary of the questions asked by the Subcommittee and that we need to address and plan to address going forward.

So where do we go from here on 16793? The staff has provided additional information to the Subcommittee on a couple of the aspects of the questions that were asked. There were some documents that we provided to the ACRS staff. There seems to have been based on some email traffic some questions as to whether the Subcommittee has actually seen a document that we provided. I asked the staff, ACRS staff --

MEMBER BANERJEE: To the Owners Group?

MR. SCOTT: There was an industry test document that you requested.

Paul, what was the name of that document, please?

MR. KLEIN: It was a paper, Mike. It was an industry document. It was requested on a Subcommittee and we provided it.

MR. SCOTT: Do you remember what it was?

MR. KLEIN: It was related to, I think, fouling, but I -- it was an old report.

MR. SCOTT: Do you have that?

MR. KLEIN: Yes.

MR. SCOTT: Okay. So that we have provided some additional information, but we have not addressed with you most of the questions that you asked. We mentioned the fact that the Owners Group plans additional testing and we need to evaluate what they tell us in response to the questions that you raised.

And the staff, as we talked about, and the Owners Group, will return to brief the Subcommittee in due course, depending on when we sort out the testing with them.

MEMBER BANERJEE: You know, the Commission has an interest in this and the question came up in our meeting yesterday, and we sort of promised them a letter. They're anxious to get the letter --

MR. SCOTT: A letter on that particular issue or 191 in general?

MEMBER BANERJEE: Well, it was a general question, but sort of the answer I gave was that most of the issues on the way to resolution and the issue which still requires some attention is this downstream effect. So I mean I'm not sure what sort of a letter we're going to give them. But they want a letter.

That's clear.

And after the meeting, Commissioner Jaczko talked to me and said we're looking to hear from ACRS. So obviously, we owe them a letter. Now exactly what we cover in that letter, I don't know, but suddenly the downstream effect being important, we need to write a letter on this WCAP or whatever the equivalent is or the full subject of downstream effects, however you guys want to treat it.

It maybe end up having to write a couple of letters, but --

MR. GROVE: This is Jack Grove. Bill and I were at the meeting yesterday, listening to the dialogue and it's pretty clear even though the last time we updated, Commissioner Jaczko, was just several months ago. GSI-191 is a project that rapidly evolves and we concluded it was clear that it was time to rebrief the Commissioner TAs on the status of 191 as well as the status of BWR strainers.

So we're scheduling that briefing now to bring all the Commissioners up to speed on the latest information. I don't know if that helps you with when you made forward letters or not --

MEMBER BANERJEE: I think it would help that you brief them, of course, but I think we owe them a letter. That's more or less the situation.

Exactly what the letter should be on, I think we should discuss as a Committee and decide how wide that letter should be or whether it should focus on a specific issue like downstream effect.

MR. SCOTT: If I might make a suggestion, you might want to table that until we talk towards the end of this presentation about what we plan to tell you and when we plan to tell it to you.

MEMBER BANERJEE: Let's go on.

MR. SCOTT: We may be in a better position to do that.

Head loss testing. The staff has questioned whether various aspect of the vendor for foreign testing is conservative and prototypical. For example, we have had concerns, as you see here in debris preparation and introduction how is the debris added to test loop and in what order is it added? Do you put the particles in first? Do you put the fibers in first? Do you throw them all in together? And you may not be surprised to find out that it matters what order you put them in.

And it turns out that particles go in first, followed by fibers debris, fine fibers debris, tends to be the worst case situation based on our observations when testing is done.

If the particles set up the fibers for an adherent thin bed and a potentially problematic thin bed and by thin I mean quite thin. Less than the one-eighth inch that we used to think of as the cut off of what a thin bed would be. It doesn't take much of the right kind of debris and the right order to cause a significant head loss.

So we have had questions about that with the vendors as we have observed the testing. Some of them put the fiber in first which we used to think was an appropriate way to do it. Now we're not so sure. Of course, the burden is on the licensees and their vendors to show us that they have a conservative protocol for their plant-specific conditions and this issue, and I can't emphasize this enough is extremely plant specific. Some plants have a little problem. Others have a significant problem.

So we have gone round and round on this. I do think that we are approaching the point, as I mentioned to you, where we're okay with the test protocols and now the licensees need to run the tests and validate that their strainer can withstand this situation.

MEMBER APOSTOLAKIS: Mike, why do some

plants have big problem and others don't?

MR. SCOTT: It has to do with how much fibrous and particulate insulation they have. Some plants and actually the NRC encouraged the industry to do this two decades ago. They said when you're going to do a modification that causes you to pull insulation out of your inside container, consider whether it makes sense to replace it with reflective metal insulation.

And reflective metal is good in some applications, maybe not so good in others, but there are plants that either started that way or have gone that way to where they have very little fiber in their plant. And some of these plants the only fibrous insulation, fibrous debris source term is what we refer to as latent debris, stuff on the floor. There's no fibrous insulation still in the plants. Those plants, we are prepared largely at this point, based on the information that's been provided to us, to conclude they have reasonable assurance that they will not experience this phenomenon.

MEMBER APOSTOLAKIS: The plant variabilities is due to the fact that some plants don't have fibrous --

MR. SCOTT: That's a major impact also.

There's a variation in what kind of buffer they use, what kind of sump buffer they use. So there are various things. But I would say --

MEMBER BANERJEE: Pump geometry --

MR. SCOTT: Yes, near-field settling refers to -- let's say -- remember that these are test regs that are in a warehouse somewhere. They're not testing in the plant. So what they do is, is they put the debris and observe as it goes to the strainer.

Well, okay, so let's say that some of the debris settles on to the floor in front of the strainer. That's fine, and they can take credit for that if that would happen in the plant. But they have to show that. They have to show that any settling in the test rig is representative of what would happen in the plant. So that's what that refers to.

MEMBER BANERJEE: It depends on how stirred-up the flow would be in the plant compared to in the test rig.

MR. SCOTT: Well, -- and they have to make sure that their flow rate is prototypical is not trivially easy --

MEMBER BANERJEE: No, not at all. In fact, last May when we first had a very interesting meeting, the Subcommittee with industry, four or five plants came and there were quite a lot of questions about the protypicality of the tests.

MR. SCOTT: Yes.

MEMBER BANERJEE: And I think they've been trying to respond to a lot of these questions and I think very good progress has been made in that direction.

MR. SCOTT: Yes, we believe so. But it's been a struggle. It's been a struggle. And what we ended up with is a number of plants had already tested under these problematic protocols from our perspective and so now we're in the mode of asking them to justify why their previously done work is adequate. That could cause additional testing, for example. So that's why we want to cleanly wrap this thing up, but it's just not clean.

MEMBER BANERJEE: And then there's only a limited number of places where you can do the testing, so the full thing is --

MR. SCOTT: That's right. There are about a half dozen vendors. The tests are significant expense for the licensees, so obviously, they don't enjoy having to retest, but at the same time they have to show that they've adequately addressed this and that their strainers will pass muster.

MEMBER BANERJEE: It's a massive job. I mean each plant is different.

MR. SCOTT: There's a lot of evaluations. There are many aspects to the evaluations. Each one is backed up by its own type of test of analysis. Yes, it's extremely off-scale high, complex issue. Yes, it is.

I think I've talked about this. I guess the last bullet, the licensees, of course, can use any approach. We don't dictate the approach. We can say okay, here's what we have observed, for example, we have observed that a thin bed composed of particles going in first, followed by fine debris, fine fiber only, is likely the most problematic situation for head loss. And then so a plant needs to consider whether that applies to their specific conditions.

Another protocol might be okay, but we'd ask questions to state that. Why? Well, one reason test that we observed. I mean I already mentioned this to you was particles followed by fine fiber and notice I put the word only in parens there. It turns out if you add more course fiber, that turns out to be better, lower head loss. Because there's more paths for the water to get through the coarser debris that sits on the strainer.

Unfortunately, for the plant though, typically, their transport analysis may show that only the fine fiber really transports. And if that's the case, then you cant' dump the big stuff on there. And so that's been a subject of some angst.

I think I've talked about the other points here. Likely to be challenging for high fiber and maybe even medium fiber plants to use this test protocol to show that they don't have a problem and that sort of case, there may come a point here in the near future where that plant needs to conclude and needs to take additional action such as additional modifications to fully address this issue.

Supplemental reviews. We are reviewing the supplemental responses that we got in February. That is our top priority right now. There will be supplements to the supplements, so to speak, for the many plants that were not done the first time through.

And we're likely to send requests for additional information to most plants. What we're trying to sort out now is how do we carry a request for additional information regarding the in-vessel issue which is, of course, still under review by the staff.

For low fiber plants, those that have

mostly RMI, as I talked about, reflected metal insulation. They are likely to see few RAIs. We are prepared to accept for several of those plants right now, that they're not going to experience a problematic thin bed, a build up of debris, on the ECCS strainer. Those plants are close to being done and that's good. Because we would rather focus our resources on the plants that are not so close to being done and so we are -- we're trying to get at the last couple of things here, notably WCAP 16793 in order to close the issue for the low-fiber plants. We're not quite there, but we're getting close.

We anticipate and I will not raise my right hand on this and swear to it, but we anticipate that we will close this thing in 2009. The last time I came and talked to you, we anticipated we will close it in '08, so this is a very, pardon the pun, fluid problem, and I warned you in advance. I warned you I was going to do that. And so while we have a goal. We just have to see what develops here.

MEMBER APOSTOLAKIS: You are consistent.

(Laughter.)

MR. SCOTT: That's not the kind of consistency I'm proud of.

MEMBER BANERJEE: It's like every

President that has said about oil independence --

(Laughter.)

It's a mirage that fades.

MR. SCOTT: To give it its due though, (a) it's a very complex issue; and (b) both the industry and the NRC has devoted huge resources to addressing this issue and enormous improvements have occurred. But to try to get to the finish line and get past the past few of these things is a real problem. And to say otherwise, I wouldn't be frank with you.

So what are we going to do?

We plan to close these issues for each plant and because it is plant specific, we're going to close it one plant at a time. And then generically. Based on what? Review of the supplemental responses which may involve supplements to the supplements. Review of region inspections of the licensee corrective actions and what I mean by that is is we have asked the regions through a temporary instruction to go validate that the licensees have done what they said they were going to do. If they said we're going to put in a certain amount of strainer, they did. If they said they were going to change a certain amount of insulation, they did. The regions are not involved directly with the review of the analyses and testing.

That's our job here. But we do take their input to have additional confidence that the licensees did what they said they were going to do.

And as we've told you in past meetings, we did a number of comprehensive audits which are actually the results of which are visible on our website. And we are looking at their responses to the audit open items.

The open items were -- the audits were a way for us to take a detailed look at certain plants representative of the various types of strainers and testing out there to increase our confidence that the issue is being addressed. Obviously, we don't have the capability to audit every plant in that way. It sends a team of eight or ten people to a plant for a week. We can't do everybody, but we did nine of them. And so those plants to be representative of, for example, one particular vendor's test protocol. And so we expect the licensees who use that vendor to have paid attention to the audit results that occurred at the other plants and we are to a limited extent validating that by going through the Generic Letter responses that have been provided to us although I would candidly say that the Generic Letter response reviews cannot be at a level of detail the same as an audit.

We will -- we'll get to the point here in all likelihood where a plant has not done all its mods. You remember I mentioned that one piece of insulation at Diablo Canyon that won't be replaced until '09. But they will have done their testing and their testing will be based on the plant configuration after the last mod goes in. There will likely come a point where we'll say that plant is done pending this particular action which we will track to completion. We don't want to drop the ball and have the plants not make the corrective actions that they plan to make. So we will track them. NRR will track them until they're all done. But we may at some point close the generic safety issue and close the Generic Letter for a given plant based on the fact that the test results are good and the mod commitments are there. So just to let you know about that. Otherwise, we couldn't close this issue until the last mod is made at the last plant and we don't currently think that's a useful way to go into this.

At the same time we have to accept the burden that we don't drop the ball after we do close it, so we're going to have a process in place to ensure that occurs, that it doesn't occur.

I mentioned to you that we talked about what we want to come back to the Committee with. We, of course, need to come back to the Subcommittee to talk about the in-vessel downstream effects. The owners group has intentions of doing their testing this month or next. Before that really occurs, at least final testing from the staff's perspective we have to agree basically buy into what they're planning to do and we haven't gotten to that point yet.

We might be in a position, we should -- I have a reasonable amount of confidence that we will be in a position in the fall, perhaps the early fall, to come in and tell you a good deal more additional information. For example, we can talk about the Diablo Canyon testing. Hopefully, the Owners Group will have already done their testing and the staff will have a view on that. I don't think in the early fall we will have a revised SE for you to look at because only at that time will we have, if the Owners Group meets their present plant, only in September will we have their revised topical report.

So there is still some months out here to do that, but I think there might be value in the fall, given the Subcommittee and potentially full Committee an update on this in-vessel downstream which currently is the last issue to be fully addressed by the staff.

We can come in and talk to you about the testing protocols and results. Of course, you got a briefing a year ago on what type of protocols were out there. We can update you on what was agreed on or what the staff reviewed and the changes that the licensees and the vendors made in order to address our concerns with those protocols so we can give you some discussion of that.

We can come in and talk to you about where we stand on review of the licensee supplemental responses. We are early in that process. You had, as I mentioned, I've already given you one top level result that we think for the low fiber plant several of them are basically done. But we're going to get into the higher fiber plants and we will come back and tell you how we're doing with that.

We will tell you, hopefully, in the fall, the final results of the chemical effects peer-review scoping analyses. I mentioned there were four issues that the Office of Research is proposing. Might need additional work. NRR plans to review that and we'll reach a conclusion on that and tell you what that conclusion was and the basis for it.

And the staff would also plan to report to you the -- some results of additional confirmatory chemical effects testing that the staff has sponsored at Argonne National Laboratory.

MEMBER ARMIJO: Could you just briefly say what kind of tests those are?

MR. SCOTT: No, I can't, but Paul Klein can.

MEMBER ARMIJO: Okay.

MR. KLEIN: Paul Klein with NRR. We had asked ANL to look at some specific things to help us in our review, the GL supplements. In particular, we asked them to do some tests with the WCAP surrogate. We asked them to look at chemical injection and how head loss with the one vendor approach would compare to head loss with the WCAP surrogate and then we also asked them to corrode aluminum in their test loop in a sodium hydroxide environment and then try to benchmark that head loss from the corrosion product in subsequent precipitation to the other two processes.

MEMBER ARMIJO: Thank you.

MR. SCOTT: So that, to answer you question earlier is what we would propose to brief you on in the fall some time. That may not be timely from the perspective of the letter that you're being asked to write. I don't know. But this was our thoughts as to what and when, in the fall, September, October, something in that nature.

MEMBER BANERJEE: I think what we need to do after your presentation is have a brief discussion as a Committee to see what sort of a letter we could write based on the information that we would have at that time.

We've noted the things that you can supply us by the all.

It seems to me that everything except downstream effects could be in a state where we can write something about. You can give us a status report on most things, right.

MR. SCOTT: In the fall, you're speaking of.

MEMBER ARMIJO: Yes.

MR. SCOTT: We'll have a much better idea about how the Generic Letter reviews are shaping up. How many plants are needing to do additional work, that kind of thing. What we decide and what they decide to do about the case where they tested under a protocol that we didn't buy into.

MEMBER ARMIJO: I think considerable progress has been made since our last letter to the Commission.

MR. SCOTT: I think so too.

MEMBER ARMIJO: So I think we should document that in some way.

MR. SCOTT: Obviously, whatever we can do to support that documentation, we'd be happy to do.

MEMBER ARMIJO: Right. I mean -- I mean I wouldn't mind waiting until everything was closed. I don't know about the rest of the Committee, but I think from what we heard yesterday, we do owe them a letter and they want the letter.

They probably will get a letter.

MR. SCOTT: Okay, that concludes my remarks for GSI-191, but I'm going to throw in one more item here.

MEMBER ARMIJO: Which also arose yesterday, Jaczko asked me about this, so --

MR. SCOTT: The long and sordid history of sump issues goes back to the 1980s, at least, probably goes back before that --

MEMBER CORRADINI: Seventies.

MR. SCOTT: Seventies, okay. As far back as I'm aware of it. The PWRs were resolved in the 1980s as a result of the information obtained from the PWRs and as a result of certain events that occurred at BWRS, both in the United States and abroad, the

Agency initiated regulatory actions in the mid-1990s. The industry took corrective actions for BWRs that made your strainers bigger. And we closed the issue for BWRs in the '90s.

As a result of the information obtained in the look at the BWRs, additional questions arose regarding PWRs and so in -- yes, one of those -- and so in 1996, GSI-191 was initiated. We have spent the last decade ironing out the issues for PWRs and we have gotten much, much smarter than we were as an agency when the BWR work was done. So that leads to the obvious question, do we need to evaluate the BWRs again?

We would like at the end of this process to achieve a consistent regulatory basis that doesn't require further rounds of Bs, Ps, Bs and Ps. So one of our objectives in looking back at the BWRs is to achieve appropriate regulatory consistency. Hopefully, we are getting to a high-enough state of knowledge on these complex issues now that it will not make sense from a cost-benefit basis, a safety basis to further pursue second and third order effects indefinitely.

Okay, so we really want to be -- we want to get a consistent regulatory basis that shows that safety is being achieved and that we can move on beyond these issues.

Why are we at different places for the BWRs and PWRs? This is probably fairly obvious to you, but just to state it, obviously, we have a different strainer design. We have a different configuration. The Bs have the suppression pools versus the sumps and the Ps. Different conditions chemical-wise. The BWRs, by and large, do not have -- they don't have chemical buffers. The PWRs do. The way the ECCS is operated obviously varies between the two reactor designs. So there are all sorts of reasons why it might be appropriate to have a different treatment for Bs and Ps. So just the fact that they're different doesn't necessarily mean that's a problem.

But at the same time we are smarter now. There are additional issues that have arisen, for example, chemical effects. That really was not necessarily addressed back in the '90s. I say necessarily because some work was done regarding impact of corrosion products from the suppression pool back when the BWRs were addressed. Were they adequately addressed, we don't know.

So we're going to go back and look at it again.

We are asking the Office of Research to evaluate the differences and recommend additional actions, if warranted, a scoping study. Research has begun that work. They just started it. Their objective is to provide us the results of that scoping study by the end of 2008. That information will help inform NRR as to whether additional regulatory actions are needed to address BWRs. However, we are not standing on that work alone. We have been encouraging the BWR Owners Group to take the initiative to address the issues themselves rather than waiting on us to come out with a multi-plan action that might be painful for the industry. We are not prepared to do that at this point, but we are encouraging the BWR Owners Group to take actions to avoid us having to get to that point.

We will consider further regulatory actions based on the results of that work. I will say that I met with the BWR Owners Group just yesterday and we received a very encouraging picture from them that they recognize that there were questions to be answered and they largely signed on to answer those. Of course, they said we don't have funding for this yet and we have to get the funding and we don't know the schedule, but we recognize additional questions needs to be answered from which the staff had sent them by the way.

We sent them a paper that said here are some specific treatment disparities that we think need to be looked at further, for example, chemical effects, in-vessel downstream effects for BRWs may be more problematic than for PWRs because of the channels that inhibit cross flow between the assemblies.

As the Subcommittee Members may recall, when we came and briefed you, it doesn't take a whole lot of flow coming into the core of the PWR to provide adequate flow because of the cross flow between the assemblies. It's a small amount of flow. For BWRs, that might be more challenging and so that's one of the points that we think they need to look into. And they agreed to look into it.

So we are encouraged, based on yesterday's meeting, it was actually a change in their position from a previous meeting we had with them in November where they didn't think certain issues needed to be addressed. I think their perception of the scope is broadened and so we were pleased with that.

So we have a plan that we are working to address the issue for BWRs and to attempt to achieve an appropriately consistent licensing basis for the two reactor types.

So we're early in that process yet and we're not prepared at this point, obviously, to talk about industry results on this, but we're embarked on the process.

MEMBER CORRADINI: Mike, we should conclude, I think.

MR. SCOTT: We will conclude with this slide. It says as you already know, as several of you have referred to, GSI-191 is a real complex issue. We are working hard to resolve it. Just for the sake of argument, would every member of the staff who has been involved in this issue, please raise his hand.

(Laughter.)

Tom Hafera, ex-member of the staff. Chen Lai Lui, NASA. They're all over the place. We have committed enormous resources for one issue to address this issue, so we are working hard to get done with it and we have had challenges and surprises throughout.

It is possible that when we're done with the testing and the analysis and we've validated the testing and analysis is okay, additional mods may be needed and that will be up to the plants to do and we will have to deal with that at that time.

As I mentioned to you, our current plan is to resolve the issue in 2009 and we are doing our utmost to get that done.

Subject to your questions.

MEMBER BANERJEE: Questions?

MEMBER MAYNARD: I don't have a question, but just to get it on the record for this meeting, I think you should identify that you, Dennis, and I went to the Salem plant and actually observed a screen that went into containment and saw the design. At least that plant had been receptive to comments made at previous meetings and it incorporated some of those. So just for the record get that on there.

MEMBER BANERJEE: right.

MR. SCOTT: Coincidentally, that was also a plant we did an audit on. So they have a number of issues to address from our audit.

MEMBER BANERJEE: So I think we've got a good update. We have to discuss at some point what we want to do.

VICE CHAIRMAN BONACA: So the BWR, it's an interesting point. Yesterday, we raised the issue of back-pressure credit for power uprates with the Commission and we're still granting back-pressure credit when this issue is still open.

MR. SCOTT: I'm sorry, say that again?

VICE CHAIRMAN BONACA: I'm saying we're still granting back-pressure credit, and yet now the issue of recirculation for BWR is reopening.

MR. SCOTT: You mean containment over-pressure? Oh, okay.

There are some PWRs that have -- some of them have quite low margins, net positive suction head margins and they have asked us not to much for post-accident pressure but for atmospheric pressure credit which we have granted. But that's an on-going discussion as well.

VICE CHAIRMAN BONACA: That's an on-going discussion.

MEMBER BANERJEE: So I think with that, I'll turn it back to you.

Thanks, Mike. Very nice update.

MR. SCOTT: You're welcome.

MEMBER BANERJEE: Very good presentation.

CHAIRMAN SHACK: Don't run away, gentlemen. That is the end of the meeting. We can go off record now. So the meeting is adjourned.

(Whereupon, at 11:44 a.m., the meeting was concluded.)

CERTIFICATE

This is to certify that the attached proceedings before the United States Nuclear Regulatory Commission in the matter of:

Name of Proceeding: Advisory Committee on Reactor Safeguards

Docket Number: n/a

Location: Rockville, MD

were held as herein appears, and that this is the original transcript thereof for the file of the United States Nuclear Regulatory Commission taken by me and, thereafter reduced to typewriting by me or under the direction of the court reporting company, and that the transcript is a true and accurate record of the foregoing proceedings.

Charles Morrison
Official Reporter
Neal R. Gross & Co., Inc.

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

US-APWR
Standard Design Certification Project Overview

To:
Advisory Committee on Reactor Safeguards

By:
Jeff Ciocco, Senior Project Manager
U.S. Nuclear Regulatory Commission

June 6, 2008

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United States Nuclear Regulatory Commission
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US-APWR Application Status

- Pre-application review meetings began July 2006.
- Topical Report submittals began January 2007.
- Received Mitsubishi Heavy Industries (MHI), LTD, US-APWR standard design certification (DC) application on December 31, 2007.
- Acceptance review completed and docketed application on February 29, 2008. (Docket Number is 52-021).
- Phase 1 licensing review underway, preparing Preliminary Safety Evaluation Report and issuing RAIs.
- Luminant selected the US-APWR technology for Comanche Peak Nuclear Power Plant Units 3 & 4.
  - COL application expected September 2008.

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US-APWR Design Certification Review Schedule

Phase	Name	End date
Phase 1	Preliminary Safety Evaluation Report (SER) and Request for Additional Information (RAI)	June 2009
Phase 2	SER with Open Items	March 2010
Phase 3	ACRS Review of SER with Open Items	June 2010
Phase 4	Advanced SER with No Open Items	May 2011
Phase 5	ACRS Review of Advanced SER with No Open Items	August 2011
Phase 6	Final SER with No Open Items	September 2011

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DCD Chapters and Topical Reports

Chapter Project Manager	DCD Chapter	Topical Reports (SER Dates)
Mike Takacs	Ch 10 – Steam & Power	
Ngola Otto	Ch 11 – Radioactive Waste Management Systems	
Ngola Otto	Ch 12 – Radiation Protection	
Mike Takacs	Ch 13 – Conduct of Operations	
Ngola Otto	Ch 14 – Initial Test Programs	
Mike Takacs	Ch 15 – Transient & Accident Analyses	- Non-LOCA Methodology (05/2009) - Large Break LOCA Code Applicability (05/2009) - Small Break LOCA Methodology (04/2009)
Peter Hearn	Ch 16 – Instrumentation & Controls	
Jeff Ciocco	Ch 17 Quality Assurance & Reliability Assurance	- Quality Assurance Program (QAP) Description for Design Certification (01/2008)
Steve Monarque	Ch 18 – Human Factors Engineering	- HFE Process & HSI System Design (09/2008)
Jin Chung	Ch 19 – PRA & Severe Accidents	

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Status and Path Forward for Generic Safety Issue 191, Pressurized Water Reactor Sump Performance

Presented by:
Michael Scott
Office of Nuclear Reactor Regulation
Presented to:
Advisory Committee on Reactor Safeguards
June 6, 2008

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Background

- Generic Safety Issue 191 involves performance of PWR emergency core cooling and containment spray systems in recirculation mode in the presence of debris after a loss-of-coolant accident/high-energy line break
- Generic Letter 2004-02 requested licensees, by end of 2007, to:
  - Determine plant-specific debris generation and transport
  - Make needed modifications to show compliance with regulations in presence of plant-specific debris loading

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Current Status of GSI-191

- Essentially all PWRs have installed much larger sump strainers
- Many have done other modifications (e.g., removed insulation or replaced sump buffer)
- Fort Calhoun implementing water management initiative
- Staff and industry believe risk of strainer clogging reduced significantly
  - Significant uncertainties remain
  - Plants can continue to operate safely for same reasons as stated in GL 2004-02
- Integrated head loss testing (including chemicals) ongoing
  - Staff reviewing and commenting on protocols
  - Staff observing and commenting on representative tests intended to show adequate strainer function

Current Status (Continued)

- Most licensees received additional time beyond 12/31/07 to complete certain corrective actions
  - Downstream effects analyses
  - Integrated head loss testing
  - Plant modifications
- Most extensions for a few months; a few into 2009
- All plants submitted supplemental responses to GL 2004-02 in February/March 2008 (incomplete responses for plants with extensions)

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Chemical Effects

- Many plants did not complete integrated head loss testing with chemical effects by end of 2007
- Completion delayed by:
  - Late recognition by industry of difficulty of the issue
  - Limited number of testing vendors, requiring queuing
  - Challenges resolving staff issues with chemical effects topical report
  - Staff issues with testing methods used or planned by test vendors
- Staff issued safety evaluation (SE) on chemical effects topical report in December 2007

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Chemical Effects Peer Review

- Staff screened peer review issues in 2007 to identify those warranting further evaluation
- Office of Nuclear Regulatory Research commissioned study of aspects that earlier staff review could not disposition
- Staff currently reviewing study results
- Likely result is need for additional consideration of some of these effects
- Will report to Committee on this later in 2008

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Downstream Effects

- Ex-vessel (pumps, valves, etc.)
  - SE on ex-vessel downstream effects topical report issued December 2007
  - Some licensees have requested extensions to complete these analyses
- In-vessel (core flow blockage)
  - Received topical report WCAP-16793-NP June 2007
  - Draft SE issued in March 2008
  - Met with ACRS Thermal-Hydraulics Subcommittee March 19
  - Subcommittee had questions and concerns
  - Staff and PWR Owners Group working to address issues
  - Will return to Subcommittee as soon as issues resolved
  - Description of method in draft WCAP and some preliminary NRC staff conclusions discussed in backup slides

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ACRS T/H Subcommittee Questions and Concerns

- Flow resistance at the core inlet or first spacer grid as a consequence of deposits (maximum loss permitted and whether that could occur)
- Temperature at the screen vs. that at the core inlet and its effect of solubility of chemical compounds
- More information on local subchannel blockage and its potential for temperature hot spots
- Bypass testing and assumptions
- Driving head for flow into the core
- Potential for and consequences of debris inhibiting boric acid mixing

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Path Forward on WCAP-16793

- Staff has provided additional information to the subcommittee that may address some aspects of these questions
- PWR Owners Group plans additional testing to reduce uncertainty regarding potential for blockage at core inlet
- Staff needs to evaluate responses being developed by PWR Owners Group
- Staff and PWR Owners Group plan to return to brief subcommittee
- Timeline dependent on completion of adequate Owners Group-sponsored testing and/or evaluation

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Head Loss Testing

- Staff has questioned whether various aspects of the licensee-sponsored vendor-performed head loss testing are conservative or prototypical
  - Debris preparation and introduction
  - Near-field settling
  - Thin bed testing
- Staff's questions and concerns have had impacts on licensee test schedules
- Staff has found that most vendors now have conservative protocols – though some licensees completed testing under previous protocols with which staff has had concerns
- Licensees can use any approach that they can show to be conservative or prototypical

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Head Loss Testing (Cont'd)

- One recent test of a uniform flow strainer conducted by adding full particulate load followed by sufficient fine fiber (only) to create a thin debris bed resulted in high head loss without chemicals
- Challenge for licensees is to develop conservative or prototypical, but not excessively conservative, test protocol
- Potentially challenging for high-fiber and maybe for medium-fiber plants

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GL Supplemental Response Reviews

- Staff has begun review of supplemental GL responses
- Because of extensions, many licensees will need to submit an additional response
- Likely to send requests for additional information (RAIs) to most plants
  - For low-fiber plants, few RAIs – maybe limited to in-vessel downstream effects
- Result is final closure in 2009

Closing GL 2004-02 and GSI-191

- Staff plans to close these issues for each plant based on:
  - Review of licensee supplemental responses
  - Results of Region inspections of licensee corrective actions
  - Review of licensee responses to audit open items (as applicable)
- If a plant has not completed all modifications but has a satisfactory strainer evaluation in place and a specific plan for completing remaining modifications, staff plans to close the GL and GSI for that plant
- Staff will track all corrective actions to completion at all plants

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Subjects Proposed for Future ACRS Review

- In-vessel downstream effects
- Integrated head loss testing protocols and results
- Results of staff review of licensee supplemental responses
- Results of chemical effects peer review scoping analyses
- Results of additional confirmatory chemical effects testing at Argonne National Laboratory

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Disparities in Treatment for PWRs and BWRs

- BWR strainer issues resolved in 1990s
- For various reasons, treatment of debris-induced clogging issues has varied for PWRs and BWRs
  - Different strainer, ECCS, and core designs
  - Issues addressed at different times and based on different states of knowledge
- Learned a lot from PWR work – applicable to BWRs?
- NRR has sent User Need to ask RES to evaluate differences and recommend additional actions if warranted – RES has begun work
- Encouraging BWR Owners Group to take initiative to address potential issues
- Will consider further regulatory actions based on BWROG and RES activities

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Conclusions

- GSI-191 remains an extraordinary complex and difficult issue to resolve
- Licensees have made substantial progress in reducing vulnerability to strainer clogging and related issues
- Additional modifications may be needed (e.g., remove problem materials from containment) if licensees cannot show success in the near future with conservative testing and evaluation
- Staff expects issue resolution in 2009

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Backup Slides

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WCAP-16793 Approach to In-vessel Effects

- Limit on the maximum temperature of fuel clad is established based upon a conservative value that prevents fuel damage (in accordance with 10 CFR 50.46)
- Industry-recognized models for deposition of solids and calculation of temperature increases based on heat transfer coefficients are used
- Flow simulation code (WCOBRA/TRAC) is used to assess limit on flow reduction and still achieve adequate core cooling
- Entire chemical effects source term from topical report WCAP-16530 assumed to be available for deposit on core surfaces

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Approach to In-vessel Effects (Cont'd)

- Size and quantity of fibrous material entering the lower core region is estimated from the containment sump screen dimensions and plant fiber bypass tests
- Deposition of this material on the lower core plate, leading to flow blockage, is assessed
- Particulate and fibrous matter that passes through the lower core plate is evaluated for local flow blockage and deposition effects
- Thickness of fuel deposits (oxide + crud + chemical deposit) formed is calculated using LOCADM based on fuel decay heat, the mass of materials present, and the core surface area

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Licensee Use of WCAP-16793

- Licensees are likely to take credit for WCAP-16793-NP as bounding for their plants in showing that in-vessel downstream effects will not cause unacceptable impacts on the fuel
- Application of WCAP-16793-NP is to be in accordance with conditions and limitations contained in the NRC SE (when published)
- Licensees are expected to verify that the assumptions in the WCAP-16793-NP methods are conservative with respect to their individual plants
- Licensees may choose to develop and substitute plant-specific data, such as debris content, chemicals, strainer efficiency, etc.

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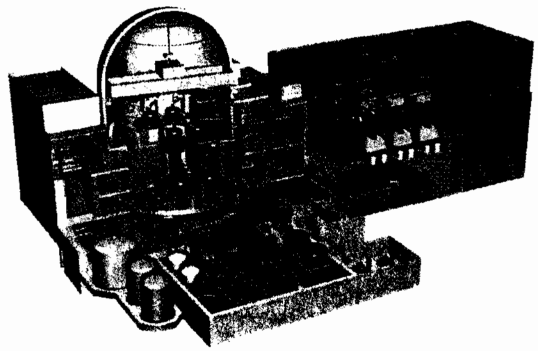
Staff Review of WCAP-16793

- Staff noted a number of conservatisms in WCAP-16793
  - Most of core entrance assumed blocked with debris – flow still adequate
  - Assumed buildup of debris on core surfaces conservative
  - Thermal conductivity value conservative
  - Worst-case local heating well below limit
  - Chemical source term assumptions conservative
  - Large margin between the chemical deposit predicted for a high-fiber plant with large amounts of calcium silicate insulation and the amount of deposit that would cause the maximum peak clad temperature to exceed the acceptance criteria

MITSUBISHI **US-APWR**

Opening Remarks
What is MHI and MHI Commitment



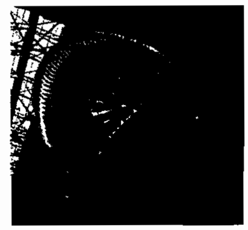
June 6, 2008

Kiyoshi Yamauchi
Executive Officer, Senior Vice President,
Nuclear Energy Systems Headquarters

Contents

2. MHI Technologies (1/6)
-Total Plant Capability-

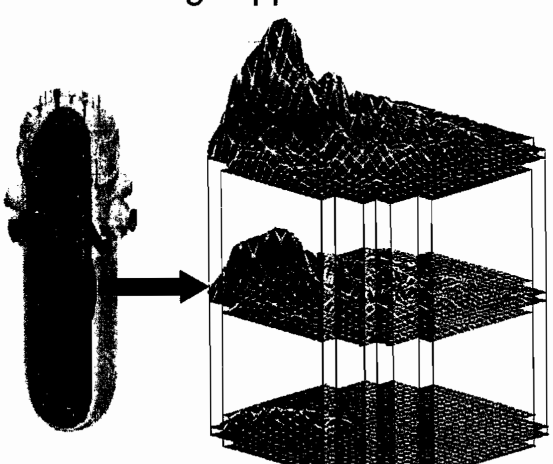
- ➢ **Total Plant Capability with "Single Point Responsibility"**
  - ✓ R&D, design and engineering, manufacturing, construction, maintenance services, and fuel supply



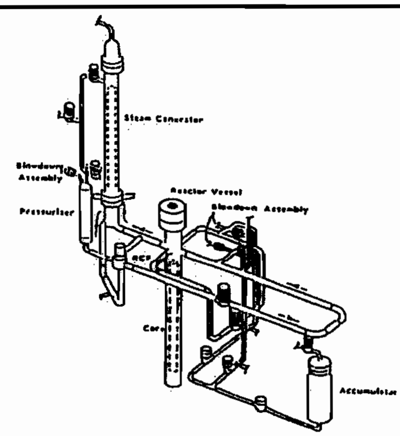
- ➢ **Globalized Quality Assurance**
  - ✓ Supporting export of nuclear components, e.g., steam generators, reactor vessels, reactor vessel heads or turbines …

2. MHI Technologies (2/6)
-Reactor Core Design & Safety Analysis-

- ➢ **State-of-the-Art Reactor Core Design and Safety Analysis**
  - ✓ Advanced analytical program
  - ✓ Verification using demonstration test facilities
  - ✓ Licensing support



Power distribution after the rod ejection from 3-D calculation

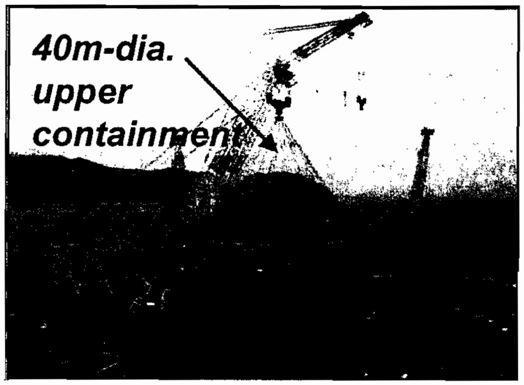


A demonstration test facility for the LOCA analysis methodology

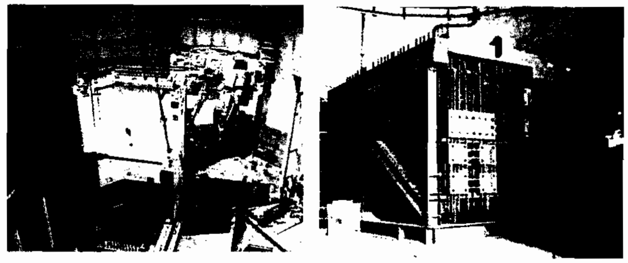
2. MHI Technologies (5/6)
-Plant Construction-

US-APWR

➢ **Various On-Site Work Reduction Techniques**



- ✓ Super large-capacity cranes
  (On-site containment Welding and formation)
- ✓ Comprehensive coordination of civil & construction work



➢ Module Utilization
- ✓ Internal structures using SC (-Left)
  (Steel plate reinforced concrete)
- ✓ Large prefabricated blocks (-Right)

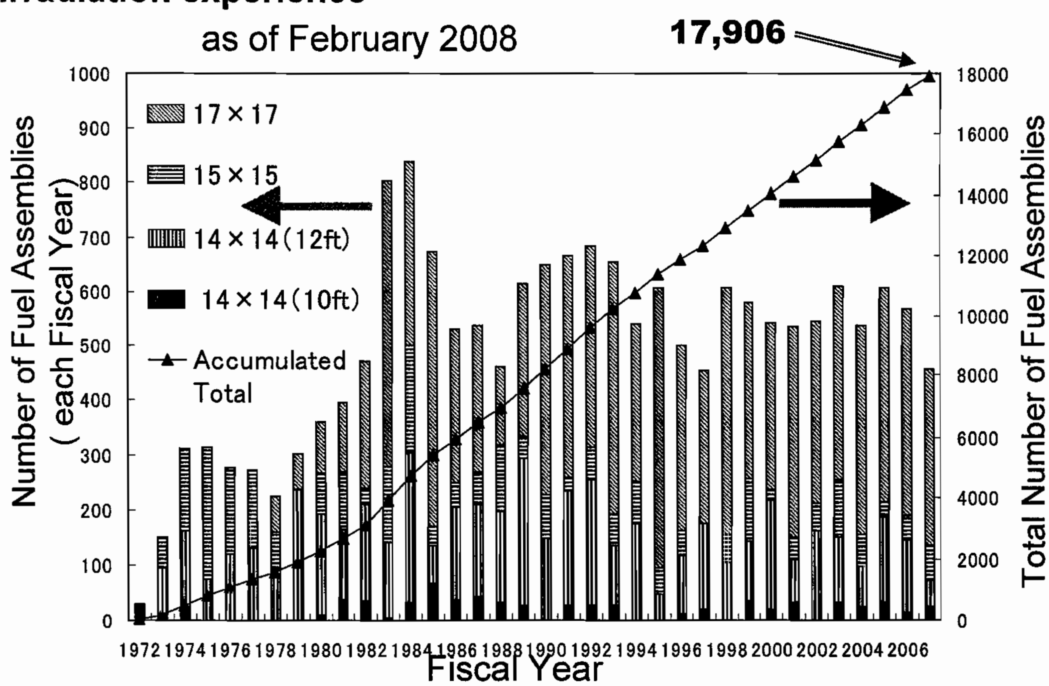
Typical achievements
(1st Concrete to fuel loading)

2 loop	: 34.5 months
3 loop	: 37.5 months
4 loop	: 40.0 months

2. MHI Technologies (6/6)
-PWR Fuel Supply-

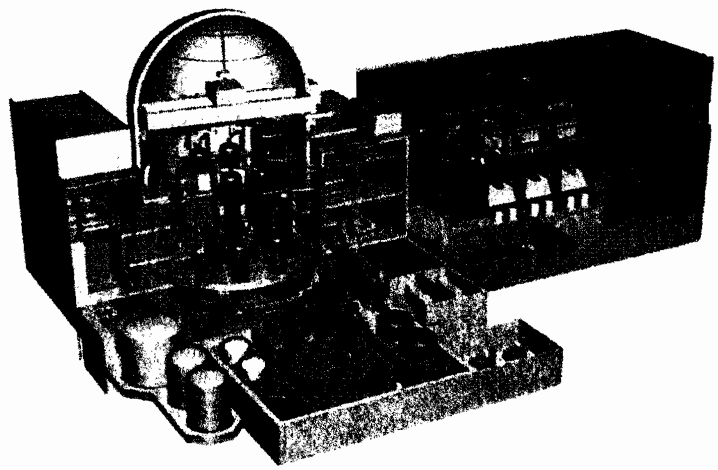
US-APWR

➢ **Leading edge technology based on abundant manufacturing / irradiation experience**



MITSUBISHI **US-APWR**

DESIGN FEATURES

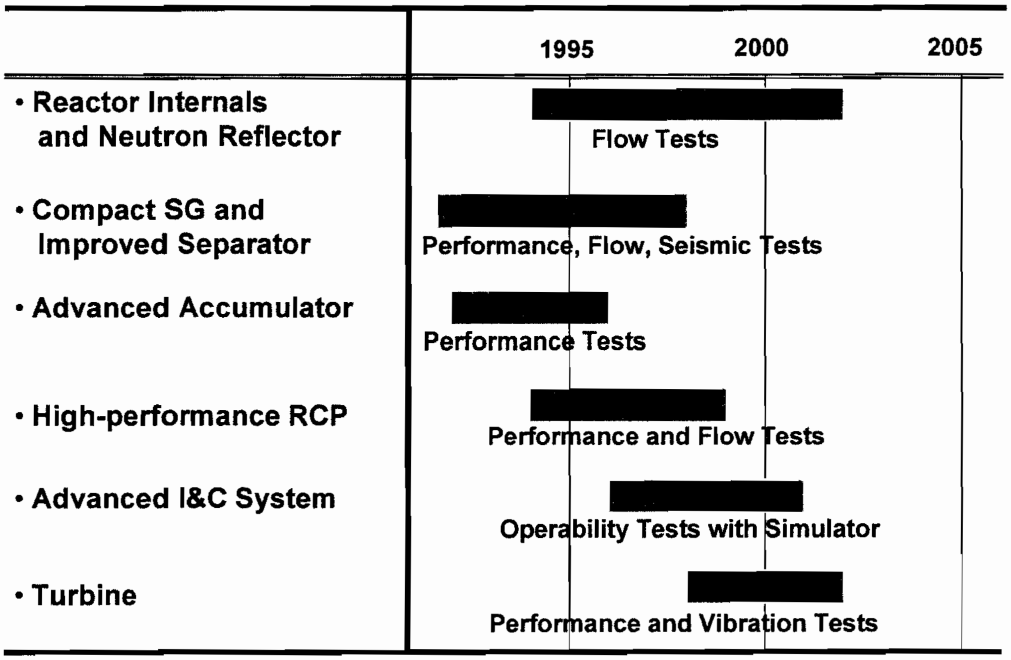


June 6, 2008

Contents

Proven, Fully Tested Technologies

Verification of Advanced Features for APWR



Reactor Flow Test

SG Separator Test

LP Turbine Test

Comparison of Output & Main Components

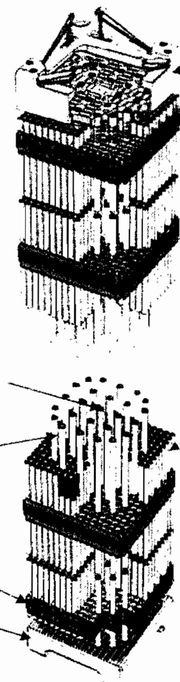
		US Current 4 Loop	APWR	US-APWR
Electric Output		1,180 MWe	1,538 MWe	1,700 MWe Class
Core Thermal Output		3,411MWt	4,451 MWt	4,451 MWt
Steam Generator	Model	54F	70F-1	91TT-1
	Tube size	7/8 in.	3/4 in.	3/4 in.
Reactor Coolant Pump	Model	93A-1	100A	100A
Turbine	LP last-stage blade	44 in.	54 in.	70 in. class

- APWR
  - ✓ 1,538 MWe output is achieved by large capacity core and large capacity main components such as SG, RCP, turbine, etc.
- US-APWR
  - ✓ 1,700 MWe class output is achieved by a 10% higher efficiency than APWR.
    - Same core thermal output as APWR
    - High-performance, large capacity steam generator
    - High-performance turbine

2. Fuel and Core Design

Fuel Assembly

- Flexible Operation
- Enhanced Economy
- Improved Reliability



17x17 Fuel rod array

14 ft Fuel active length ●

In-core instrumentation guide tube

Control rod guide thimble

Bottom grid spacer

Bottom nozzle ●
Anti-debris design with built-in filter

Top nozzle

Top grid spacer

Intermediate grid spacer
- High DNB performance design ●
- Shorter grid spacing with 11 grids ●●
- Fretting resistant spring ●
- Zircaloy-4 ●

Fuel rod
- Higher density pellet (97%TD) ●
- Corrosion resistant cladding ● Material (ZIRLO™)
- Higher gadolinia content ● pellet (10wt%)
- Large plenum volume ●
- Lower power density ●

Fuel Design

	US Current	APWR	US-APWR
Fuel Assembly			
Fuel Rods Array in Fuel Assembly	17 x 17	17 x 17	17 x 17
Number of Fuel Rods per Fuel Assembly	264	264	264
Number of Control Rod Guide Thimbles	24	24	24
Number of in-core Instrumentation guide tube	1	1	1
Number of Spacer Grids	8 / 10	9	11
Fuel Rod			
Outside Diameter	0.374 in.	0.374 in.	0.374 in.
Cladding Thickness	0.022 in.	0.022 in.	0.022 in.
Active Fuel Length	12 ft / 14 ft	12 ft	14 ft
Enrichment	Max. 5 wt%	Max. 5 wt%	Max. 5 wt%
Gadolinia Content	Max. 8 wt%	Max. 10 wt%	Max. 10 wt%
Pellet Density	95 % TD	97 % TD	97 %TD
Material			
Cladding	ZIRLO™	MDA / ZIRLO™	ZIRLO™

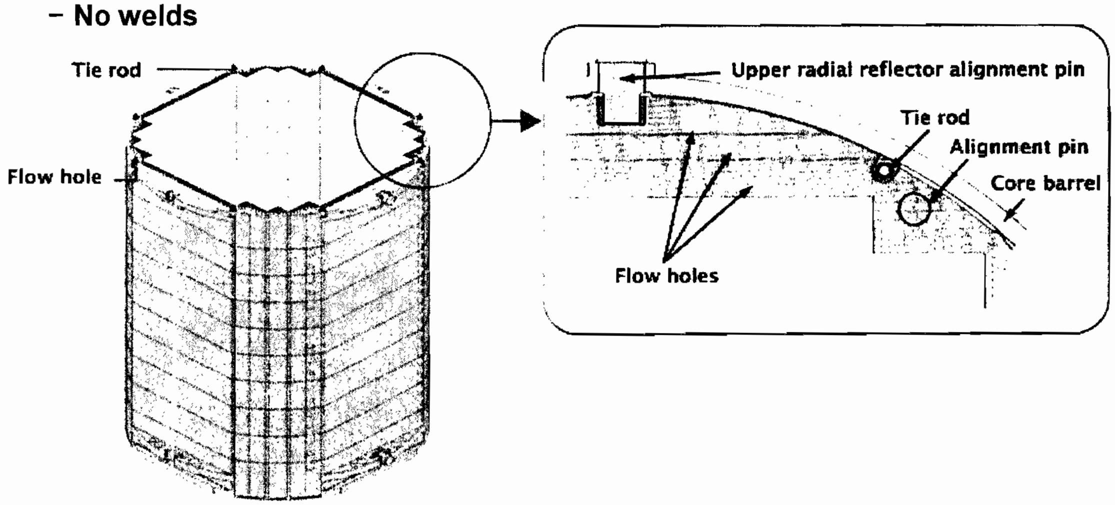
Neutron Reflector

Improved reliability

- **Significantly simplified and reliable structure**
  - Number of bolts reduced significantly and located only at out of core region
  - No welds

Enhanced performance

- **Reduced neutron exposure rate**
  - 1/3 of current 4 loop design without neutron shield



Methodology and Codes

➢Fuel Design

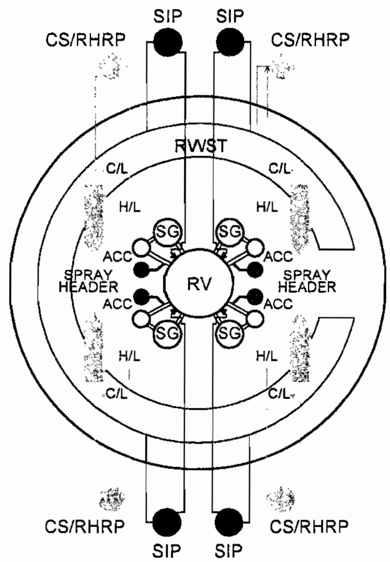
✓ FINE

- Fuel rod design code developed by MHI
- Significant post irradiation examinations and out-of-pile test
- Topical report on verification and applicability to US-APWR fuel is under NRC review

✓ FINDS

- Fuel assembly seismic analysis code developed by MHI
- Topical report on verification and applicability to US-APWR fuel is under NRC review

ECCS and CSS/RHRS

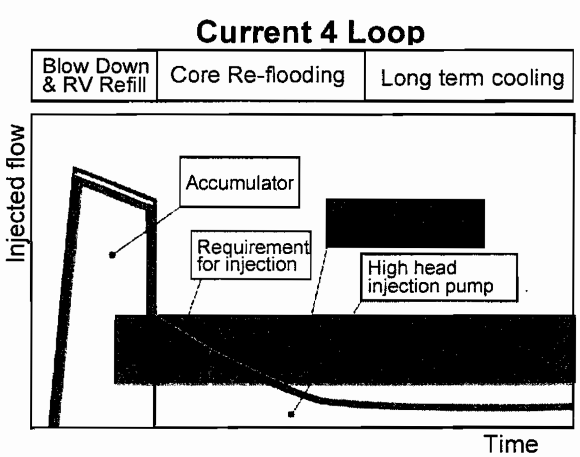


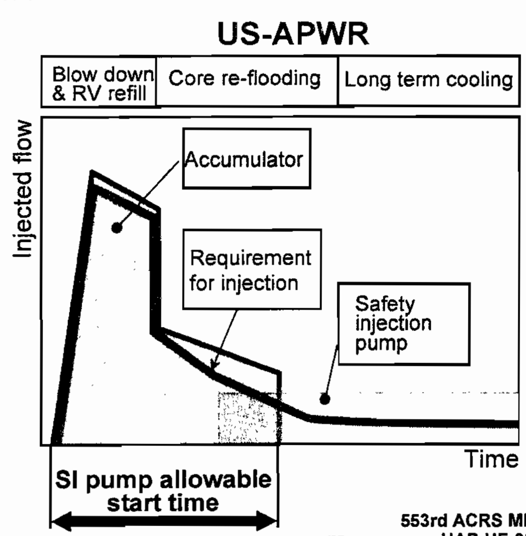
- **High Reliability**
  - ✓ 4 train configuration (50% x 4 for large break LOCA)
  - ✓ In-containment RWSP (eliminate recirculation switchover)
- **Simplification**
  - ✓ Advanced accumulators (ACC) (Integrated function of low head injection system)
  - ✓ ECCS train includes an accumulator and high head injection system
  - ✓ Direct vessel injection (no inter-connection between trains)
  - ✓ Common use of CSS and RHRS

ECCS and CSS/RHRS (cont.)

Advanced Accumulator

- ✓ Passive switching of injection flow rate
- ✓ Integrated function of low head injection system
- ✓ Long accumulator injection time allows more time for safety injection pump to start (allows use of gas turbine generator for EPS)
- ✓ Topical report on Advanced Accumulator is under NRC review

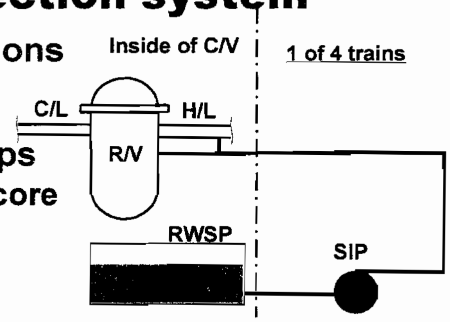




ECCS and CSS/RHRS (cont.)

Design feature of high head injection system

- ➢ 4 independent trains without interconnections between trains
- ➢ Sufficient capacity of safety injection pumps
  - ✓ Meets the safety injection requirement for core reflooding stage

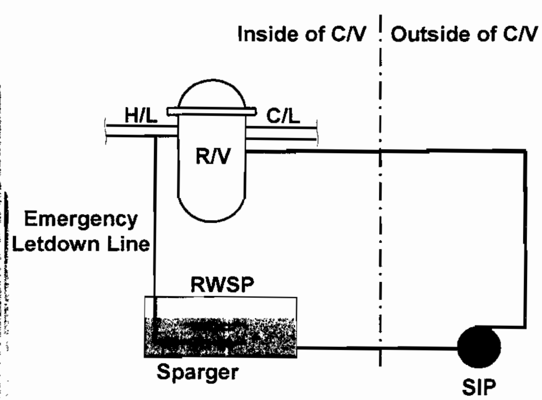


Item	US Current 4 Loop	US-APWR	Reason and/or Advantage
Trains	2 trains	4 trains	•Enhanced reliability •Achieve OLM under single failure
High head Injection	Loop injection 2 SIP + 2 CH/SIP	DVI 4 SIP	•No interconnection between trains
Refueling Water Storage Pit	Outside CV	Inside CV	•Eliminate recirculation switchover

ECCS and CSS/RHRS (cont.)

Feed & Bleed for Boration to Achieve Safe Shutdown

➢ Design Features

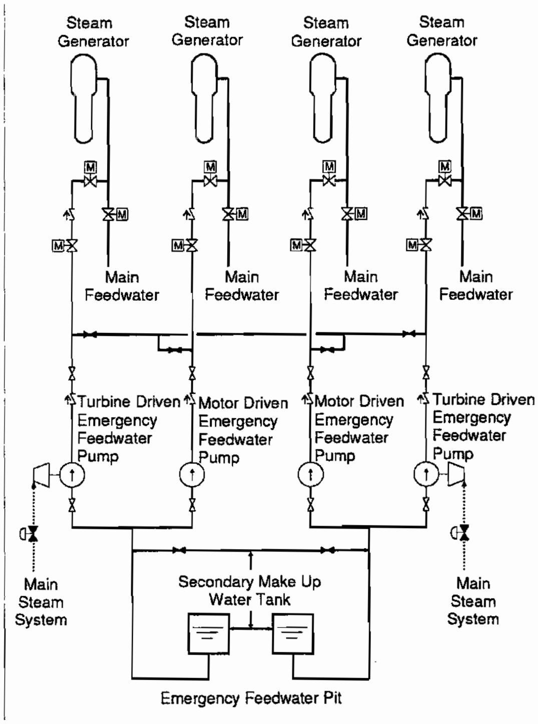


- ✓ Emergency Letdown Lines are installed from H/L to RWSP
- ✓ In Safe Shutdown operation, emergency boration source is RWSP
- ✓ The borated water is injected by Safety Injection pump
- ✓ The volume control of RCS is achieved by Feed & Bleed with SIP and Emergency Letdown Line

Emergency Feedwater System

➢ Design concept of the EFWS
- ✓ Achieve high reliability with simplified systems
- ✓ Allow On-Line Maintenance assuming single failure

➢ Feature of the EFWS
- ✓ Independent 4 train system
- ✓ 2 safety grade water sources
- ✓ Diverse power sources for the pumps
- ✓ Cross connections in the inlet and outlet of the pumps (normally isolated)



Emergency Feedwater System (cont.)

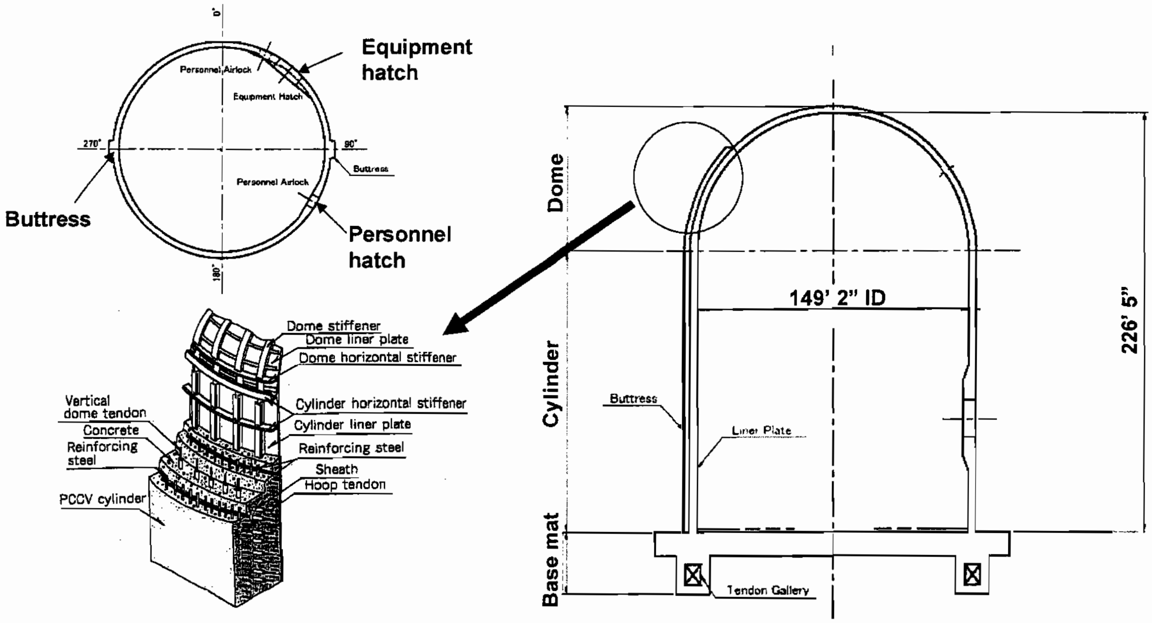
➢ 4 train configuration
- ✓ 4 pumps with diverse power sources
  - 2 motor-driven emergency feedwater pumps (50% x 2)
  - 2 turbine-driven emergency feedwater pumps (50% x 2)
- ✓ Cross connected discharge of the pumps allows On-Line Maintenance (OLM)

➢ 2 safety grade independent feedwater sources
- ✓ Two emergency feedwater pits (50 % x 2)
- ✓ Cross connected inlet of the pumps backs up each feedwater source

Item	US Current 4 Loop	US-APWR	Reason and/or Advantage
System Configuration	2 trains	4 trains	A pump is allowed OLM under the single failure
Emergency Feedwater Pump	M/D EFWP: 2 T/D EFWP: 1	M/D EFWP: 2 T/D EFWP: 2	Diverse power sources
Emergency Feedwater Source	1	2	2 independent pits (backup available)

PCCV

➢ Robust and reliable Pre-Stressed Concrete Containment Vessel with steel liner is used in US-APWR



Methodology and Codes for Safety Analysis

➢Large Break LOCA

- ✓ WCOBRA/TRAC code and ASTRUM methodology
- ✓ Approved by NRC
- ✓ US-APWR design features modeled:
  - Advanced Accumulator
  - Direct Vessel Injection
- ✓ Topical report on applicability to US-APWR is under NRC review

➢Small Break LOCA

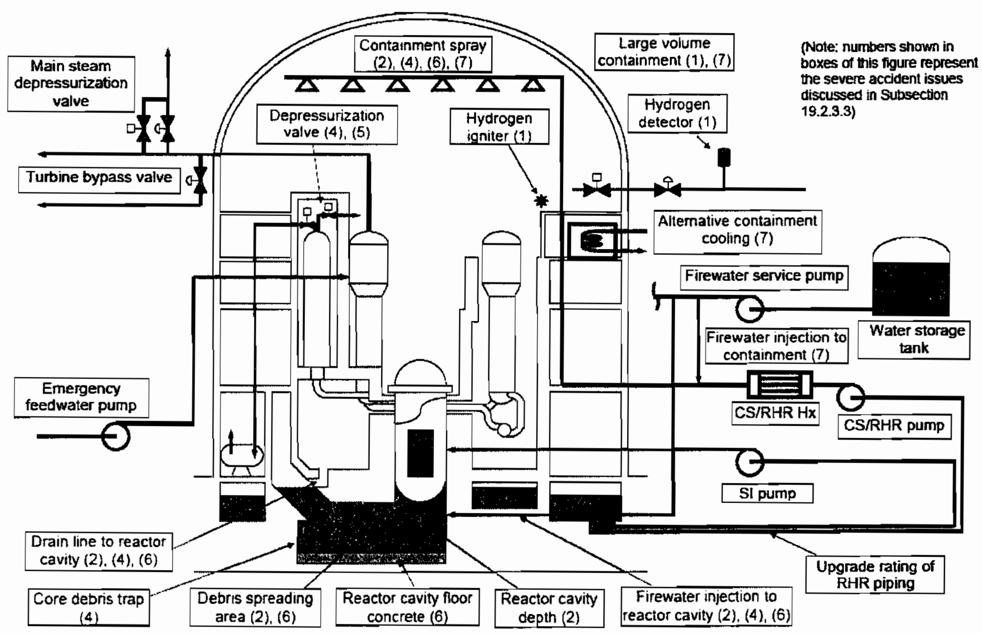
- ✓ Appendix-K version of M-RELAP5 code
- ✓ Equivalent to RELAP5/MOD3.2 widely used in US
- ✓ US-APWR design features modeled:
  - Advanced Accumulator
  - Direct Vessel Injection
- ✓ Topical report on applicability to US-APWR is under NRC review

Countermeasures for Severe Accident

- US-APWR achieves higher safety to comprehensively address severe accident and mitigate consequences
  - ✓ Demonstrate compliance with current NRC regulations including TMI requirements for new plants
  - ✓ Demonstrate technical resolution of the applicable unresolved safety issues (USI), and the medium and high-priority generic safety issues (GSI) discussed in NUREG-0933

Countermeasures for Severe Accident (cont.)

- Severe Accident Mitigation Features



Addressed severe accident issues

(1) Hydrogen generation and control

(2) Core debris coolability

(3) Steam explosion

(4) HPME

(5) TISGTR

(6) MCCI

(7) Long-term containment overpressure

(8) Equipment survivability

Overall I&C System Architecture

- Microprocessor based digital technology for most plant I&C (no electro-mechanical relays)
- Complete four train redundancy for safety I&C with each division in separate fire zone
- Distributed architecture for non-safety I&C with redundancy
- Fully multiplexed and duplicated signal transmission networks from local areas to I&C equipment rooms and Main Control Room, and between I&C systems
- Common digital platform for safety and non-safety I&C
- Diverse Actuation System based on analog technology
- Fully computerized Main Control Room

History of MHI Digital Application

- Non-safety Application History
  - ✓ Development began in 1985 with initial goal of non-safety applications and long term goal of safety applications
  - ✓ Platform originally developed in compliance with US standards, including communications independence (cyber security)
  - ✓ First installation for non-safety systems / components
  - ✓ Average 10 years operation for five operating plants
  - ✓ Applied to all non-safety I&C, 50 applications per plant
  - ✓ Over 20 million hours total operating experience
  - ✓ No unexpected shutdown caused by I&C
  - ✓ No system malfunction caused by S/W or H/W failure

HSI Verification & Validation

- ➢ HSI Verification & Validation is being conducted with U.S. operators
- ➢ Dynamic validation will be performed using Full-Scale Simulator with 8 U.S. operating crews
  - Performance Check
  - Review and Comment
  - Normal and accident scenarios
  - Normal and degraded HSI conditions

  ✓ Established Standard Design Specification

  ✓ Results will be issued as a technical report this year.
- ➢ NRC Staff visited MEPPI on June 4th.

  Demonstrated plant operation using the simulator.

5. Conclusions

- ➢ **US-APWR design is similar to the Japanese APWR currently in the stages of licensing review**
- ➢ **US-APWR is 1,700 MWe class large NPP based on MHI proven, advanced technology to improve reliability and enhance safety**
- ➢ **US-APWR meets U.S. utility's requirements and provides enhanced safety with features that address R.G. 1.206**