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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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U.S. EPR SUBCOMMITTEE

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TUESDAY

APRIL 20, 2010

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ROCKVILLE, MARYLAND

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

SUBCOMMITTEE MEMBERS PRESENT:

DANA A. POWERS, Chair

HAROLD B. RAY

MICHAEL T. RYAN

WILLIAM J. SHACK

JOHN W. STETKAR

NRC STAFF PRESENT:

DEREK WIDMAYER, Designated Federal Official
SURINDER ARORA
JASON CARNEAL
FRED FORSATY
SHANLAI LU
JOSEPH COLACCINO
TARUN ROY
JOHN WU
TIMOTHY STEINGASS
CHANG-YANG LI
JOEL JENKINS
NEIL RAY
GREG MAKAR
HANH PHAN
TODD HILSMEIER
MALCOLM PATTERSON
SARA BERNAL
ED ROACH

ALSO PRESENT:

GREG GIBSON
MARK FINLEY
DALE MATTHEWS
GENE HUGHES
MARK HARVEY
RICHARD SZOCH
CHARLES TALLY
JOSH REINERT
TIM KIRKHAM
PEDRO PEREZ
ROB PORCHET
JIM AUGUST

A G E N D A

P R O C E E D I N G S

8:45 a.m.

Introduction

CHAIRMAN POWERS: The meeting will now come to order. This is a meeting of the Advisory Committee on Reactive Safeguards, U.S. EPR Subcommittee. I'm Dana Powers, chairman of the Subcommittee. EPR's members in attendance are Bill Shack, John Stetkar, Harold Ray. Sanjoy Banerjee is supposed to join us, but I haven't seen hide nor hair of him.

Michael Ryan will be joining us for the afternoon session when we talk about radiation protection, and since we've deliberately had to accommodate the schedule for Mike, I will be inviting all participants to come with some onerous task for Mike to undertake, in payment for adjusting the schedule.

The purpose of the meeting is to continue our review of the SER, with open items for the combined license application submitted by UniStar Energy for Calvert Cliffs Nuclear Power Plant Unit 3.

We will field presentations and discuss Chapter 4, the Reactor, Chapter 5, Reactor Coolant System and Connecting Systems, Chapter 12, Radiation

Protection, and Chapter 17, Quality Assurance. Of course we have been through those chapters for the design certification.

We will also during this meeting complete our discussions of Chapter 19, PRA and Severe Accidents. That is a very optimistic wording of that statement.

I perhaps should say we will continue our discussions of PRA and their accident evaluation of the SER with open items, with a design certification document submitted by AREVA NP and the U.S. EPR design. We'll do that tomorrow afternoon.

The Subcommittee will hear presentations by and hold discussions with representatives of UniStar, AREVA NP and the NRC staff, and any other interested persons regarding these matters. The Subcommittee will gather relevant information today and plans to take the results to the review of these chapters, along with other chapters reviewed the Subcommittee and our other subcommittees, to the full committee at a future full committee meeting.

MEMBER STETKAR: Easy for you to say.

CHAIRMAN POWERS: Not very easy. Yeah, we've gone through some adjustment on when we're going to make full committee meetings and what-not, and we're really looking for points of finality or points where we were running into issues that are difficult to resolve with the full committee.

Otherwise where things going kind of expected, I don't see any reason to bother the full committee with it. I think we did an excellent set of presentations on the general layout of the reactor and general futures of the reactor at the last meeting.

I don't think they need punctuated updating on every single body blow and punch that gets delivered here. But we'll adjust that as we see that, and of course all participants can have input onto that decision to put things to the full committee.

If you've got reasons, I'll be glad to listen to them. Otherwise, I'm going to try to minimize the amount of full committee meetings. I just don't see a utility of bringing them up to speed on things that are of a routine nature.

The rules for participation in today's meeting have been announced as part of the notice of the meeting previously published in the *Federal Register*. We have received one request for a member of the public to speak at today's meeting, and time has been allocated for this during our discussions of Chapter 17, the Calvert Cliffs SER.

A transcript of the meeting is being kept and will be made available as stated in the *Federal Register* notice. Therefore, we request the participants in this meeting use the microphones located throughout the meeting room in addressing the Subcommittee.

The participants should first identify themselves and speak with sufficient clarity and volume so that they may be readily heard. Copies of the meeting agenda and handouts are available in the back of the meeting room.

We do have a telephone bridge line that has been established in the meeting room today, and I understand that we have participant from UniStar and AREVA NP on the line at various times throughout the meeting.

We request the participants in the bridge line identify themselves when they speak, and to keep the telephone on mute on the times when they're just listening. Do any of the members have opening comments they'd like to make?

(No response.)

CHAIRMAN POWERS: In that case, we're going to begin with an opening discussion by Surinder Arora, who's the project manager for the review of the

Calvert Cliffs Unit 3 COLA.

NRC Staff Introduction

MR. ARORA: Good morning. My name is Surinder Arora, and I'm the lead project manager for Calvert Cliffs Unit 3 combined license application.

We are here today to present our second batch of four chapters, which are Chapter 4, 5, 12 and 17. Each of these chapters will be first presented by UniStar, to provide an overview of the chapter and also how it fits into the application.

Then it will be followed by staff's presentation for that specific chapter. Before we start with the chapter's discussion, I will go over a couple of slides here, just an overview of the project.

The first slide which is on now is the six phases. It defines the phases and target dates we have for each phase. As the first line indicates, we have just completed Phase 1 for all chapters for Calvert Cliffs for an application.

We are in Phase 2 currently, and some of the chapters, as we are sitting here in the ACRS meeting, we are Phase 3, and the target dates are shown on this slide. Next please.

This slide is the FSAR chapters grouped by completion dates, and the ACRS presentation dates. We are in Group 3B-1, consisting of Chapters 4, 5, 12 and 17. Original schedule was two days and we are trying to fit it in one day.

CHAIRMAN POWERS: This is Ryan's doing. You've got to help me on -- we've got to really put a penalty on this.

MR. ARORA: Sure.

CHAIRMAN POWERS: Figure out something for Ryan that he's not going to like to do.

MR. ARORA: Next slide. This one slide that I want to talk about general RAI, Request for Additional Information that we have issued to the applicant, and it relates to the concurrent review of the line certification, which is being done in parallel with the COLA application review.

We have issued an RAI to the applicant, making sure that applicant will be providing us with a COLA revision when there is a change to design certification. The reason for that is a lot of sections in the COL application are by reference, and we want to make sure that all the changes in the DC arena are captured in the COLA application.

So this is a generic RAI, which will be applicable to all the chapters which have some sections by reference, and this will only be closed after this is certified.

CHAIRMAN POWERS: Well, it will be open all the time. There will be some interest, Greg, when you get to your -- to understand kind of the general outlines in your FSAR and the certification, you know, how revisions are transmitted back and forth and how you're aware of it.

I don't need microscopic detail; just generally understand the time line that exists there.

MEMBER SHACK: Just what are you looking for in here. I mean I'm looking at over -- pressure protection, which you have an open item on DCD, and here it says in the COLA, you know, no departures or supplements. What will they have to do to convince you that they've, I don't know, read this DCT?

MR. ARORA: If there is an open item on the DCD, that will have to be closed, and --

MEMBER SHACK: How do they signal to you?

MR. ARORA: That will be by the revision of the COLA.

MEMBER SHACK: But since no departures or supplements now.

MR. ARORA: At this time. But if there is an open item.

MEMBER SHACK: Okay. So that becomes a --

MR. ARORA: That's what was supplemented by their revision to us, so we can make sure that the information provided in the DCD and the information provided in COL together is a complete scope of the -- that's what the review of the technical staff scoping.

Any questions on that? That was my last slide? If there are no questions, I can entertain those. Otherwise, I'm going to turn it over to Mr. Gibson for introducing his team and start with the Chapter 4 presentation.

CHAIRMAN POWERS: So you will get to your presentation, as they're disclosable. One comment I'll make and it's relatively important for the committee to understand.

In your review, those areas that just how you do it, whether you did a review of the submitted material or if anything was done by the staff, what I would say independent confirmations or visits, audits and things like that. Pretty much as usual, but you can decide a little for us. Greg?

Chapter 4, Reactor

MR. GIBSON: Thank you. Let me do an introduction, and if you could stay for the remaining three meetings that we have on the other three chapters, I'd like to do that. I'm Greg Gibson. I'm the Vice President of Regulatory Affairs for UniStar Nuclear Energy.

I'm pleased to come before you. This is the second time we've come. The first, of course, was for Chapter 8 on electrical systems. My background is I have over 35 years of experience in regulatory affairs in the nuclear industry.

I originally graduated from Georgia Tech. I have a Bachelor's degree in Physics. I have a Master's. I also have an MBA in International Business. After I graduated, I worked for Florida Power Corporation, and then I went to work for the Nuclear Regulatory Commission, where I was with the NRC for eight years, including a number of special projects post-TMI.

I then went to Southern California Edison, where I had a very enjoyable career with Southern Cal Ed, in various areas of licensing and components. I then went to South Texas project, and was the Manager of Regulatory Affairs there for the first docket in COLA.

Actually, we did it twice almost, once with GE and once for Toshiba. I then went to UniStar Nuclear Energy, where again as the Vice President of

Regulatory Affairs we have the Calvert RCOLA and three SCOLAs.

I also have been very active in the American Nuclear Society, was the chairman of the Operations and Power Division, and met most of you in that capacity at the international meetings or at the Mileon (ph) working conference.

CHAIRMAN POWERS: You still involved?

MR. GIBSON: Yes, I am. I will be at the ICAP in San Diego and also doing a presentation panel on new reactor licensing.

CHAIRMAN POWERS: Very good.

MR. GIBSON: So with that, I'm pleased to be here. If I could very quickly, Dr. Parrish you asked a question about how do we coordinate with the design certification. The design certification is getting ready to make

Revision 2.

When Revision 2 comes up, it takes approximately three months for us to take all of the changes, identify the effective pages, put them together and then submit them as a revision. It will be Revision 7 to the Calvert COLA.

The RCOLA is tentatively scheduled to be revised September 30th of this year. We won't nail that date down until they actually publish the design certification. So that is how we move together.

We do have tracking of every, what we call a licensing basis document change request, LBDCR. For every effective change, whether it's a RAI, whether it's on ours or for theirs, every particular section is tracked by an LBDCR, and we incorporate those into the next revision.

MEMBER SHACK: It's actually somewhat faster that I would have thought. Three months seems like --

MR. GIBSON: Well, we are preparing the LBDCR packages as we go, and checking them against a living version of the COLA.

MEMBER SHACK: You would have to. Thank you.

MR. GIBSON: Okay. With that, I'd like to begin the presentation. As we noted in our first one when we were here on Chapter 8, our RCOLA incorporates by reference and uses that methodology. So in large measure, you will see for Chapter 4, for example, that we have incorporated it almost entirely by reference.

To simplify our presentation today, what we'll be doing is discussing supplemental information, and cite-specific information on the departures that we're taking from the certified design.

As you noted earlier, Dr. Powers, you've had the meeting with AREVA on Chapter 4. That was back in March, and so we are here today to talk about our COLA, and we're supported by our team mates at AREVA for this presentation.

I'll be introducing Mark and Hongqing, so we will be doing that presentation. Next slide.

With that, I'd like to introduce Mark Finley, and Mark, will you give us your vitae?

MR. FINLEY: Thank you, Greg. As Greg says, I am Mark Finley, the UniStar engineering manager. Been with Constellation Energy 26 years, most of that at Calvert Cliffs site, and most oft hat in different engineering positions.

I was three years at Ginna until 2006, as the power upgrade project manager, and I was here in February in the Chapter 8 electrical presentation. Graduated from the Naval Academy, seven years nuclear Navy. I do have a PE certification in the state of Maryland, mechanical.

Slide 4, just to sort of summarize, the presentation today, as Greg said, this is -- this should be short and sweet, because we essentially incorporate by reference all of Chapter 4 from the

U.S. EPR FSAR.

I'll summarize our FSAR content. There are two SER open items I'll discuss, and then I'll hand it back to Greg for the conclusions.

Slide 5. Just to summarize the FSAR content, again Calvert Cliffs Unit 3 FSAR Chapter 4 incorporates by reference the U.S. EPR Chapter 4. We request no departures from U.S. EPR Chapter 4. We do have one exemption request, and it is to use the M5 material in the fuel cladding and the fuel assembly structure.

This again follows AREVA. AREVA also has an exemption request to use that material. We have an exemption request in Part 7 of the COLA. You'll see an item here later to incorporate reference to that exemption request in Chapter 4. The staff has asked us to do that and we will do that.

We have no additional site-specific information in our Chapter 4.

MEMBER SHACK: Are you using M5 at your current units?

MR. FINLEY: No.

MEMBER SHACK: So you have no exemption?

MR. FINLEY: That's correct. No exemption at this time.

MEMBER SHACK: The interest in using him for a ways, because that's what AREVA wants to do in their fuel. It's not an independent judgement on your part.

MR. FINLEY: Certainly, we'll review the basis for the exemption, as we roll all of the design proposals from AREVA. But yes, we're sticking with the AREVA fuel design and the recommendation to use M5. This is obviously not the first exemption request to the Commission to use M5. There is a body of operating experience with material, and we're real confident that --

MEMBER SHACK: We're working very hard to encourage them to revise the regulation so they don't have to request this exemption. It's a slow process. Now but I was interested just in if you'd looked at M5 and what you thought about the material.

MR. FINLEY: We haven't completed our reviews, but we do know that as again, it's used throughout the industry in many applications, so we're confident in the performance. We are aware of the one issue regarding growth of the structural components using M5, and we're following that closely. But we think that AREVA's on a good track there.

MEMBER SHACK: Uh-huh. There are a

variety of other claddings in some state of development. Do you follow those?

MR. FINLEY: We do, we do. Not personally. I'm not directly involved in that process, but we have a pretty strong Fuel Group within Constellation, and of course now we're backed up with our association with EDF and their experience on different fuel types. So we intend to do a very detailed review of the fuel assembly design prior to the first core load.

CHAIRMAN POWERS: We have Department of Energy undertaking this fairly dramatic program, using supercomputers to calculate fuel behavior and things like that. Does that look like it will have any utility to you at all?

MR. FINLEY: Frankly Dr. Powers, I'm not familiar with the process. Certainly, we'd be interested in any furtherance of cladding design. Throughout the industry, there are still issues with fuel failures, fuel leaks. So we would participate and support any advances.

CHAIRMAN POWERS: The statement that you're unfamiliar is one I have to echo too. I only know that it's going on. I don't know any of the details. But it's the idea that we don't have to do experiments anymore.

We only have to do bigger computers and maybe I'm skeptical in that regime, but it is a Department of Energy initiative, and I suspect that both doing the research and the Department of Energy would really welcome input from people that have to make use of these materials, and what would be useful to them.

MR. FINLEY: I understand. I'll follow up --

CHAIRMAN POWERS: Because I mean these guys are experts in computers, and I think experts really in fuel. They're certainly not users of fuel. So it's difficult for them to make judgments on what kinds of things to do that are useful, because everything is intriguing, and somebody's little niche in this world, everything is intriguing.

MR. FINLEY: I understand. Okay. Slide 5. So no specific, site-specific information in addition to Chapter 4 from the AREVA U.S. EPR Chapter 4. There are no specific COL information items at this time.

However, I will speak about one that we expect to be added to the U.S. EPR FSAR. We'll speak about that on future pages. So Slide 6. Again, to summarize the FSAR content, we do have no contentions from the Atomic Safety Licensing Board concerning Chapter 4. There are two NRC SER open items. We'll speak about those, and there are no NRC SER confirmatory items.

Slide 7, SER open items are next, and Slide 8, we list the two SER open items. The first is related to RAI 225, which is to address the plant specific surveillance of reactor internals in regard to fluence methodology. This is that COL item we expect to be added to the U.S. EPR Chapter 4, and we've sent a letter to the staff saying that once that is added, we will incorporate the requirement in Chapter 4 of our FSAR, essentially to commit to doing this benchmarking of the fluence methodology in our surveillance program.

The second NRC SER open item, response to RAI 226, which is to include a discussion in our Chapter 4 regarding the exemption request to use M5 material. We have it in Part 7 of the COLA, but it's not in Chapter 4.

So they've asked us to put it in Chapter 4. We'll do that. We've sent a letter to the staff stating that. Questions on the SER open items?

(No response.)

MR. FINLEY: Slide 9, then, is conclusions. Back to Greg.

MR. GIBSON: Yes. As we said when we started, the beauty of the Part 52 process is our ability to reference the certified design, and we have obviously demonstrated that today. We thank you for your patience. Again, we have no SER confirmatory items and the two SER open items, again we just recently gave those. We just wanted -- we put the dates in so you'd recognize this as the information we just provided to them, and we appreciate the staff.

We know they haven't had a chance to roll on those. But we're very confident that those items will be closed out very quickly. So with that, we appreciate the opportunity to come before you.

CHAIRMAN POWERS: Bill, what are we, on this fluence materials benchmarking? We were designing this reactor for 60 years.

But I mean let's face it, it's probably going to be -- I mean if it's a good reactor, at year 59 we'd probably want to continue it. Now would we have -- can we do analysis specific of materials embrittlement --?

MEMBER SHACK: We can certainly extrapolate.

CHAIRMAN POWERS: It's extrapolation.

MEMBER SHACK: We asked that question, as you recall, with the EPR, as to whether their surveillance program was planned to go beyond 60 years and it wasn't, which apparently does meet the regulations. But it just seemed to me a little short-sighted. Maybe that's really a question back again to the person who's actually going to own this thing, and presumably would like to operate it for as long as he can.

Whether, you know, you're looking at a surveillance program that takes you out beyond the current design life.

MR. FINLEY: That's a good point. In fact, at this point, we haven't established a surveillance schedule to support a lifetime greater than 60 years.

But as part of developing the program for surveillance of reactor vessel materials, we will look at developing a schedule that would support extending the license, yes.

CHAIRMAN POWERS: What I'm fishing for, without asking the question, is your own thinking. How long is this reactor good for?

MR. FINLEY: Certainly, we've gained,

since the first generation of operating reactors in this country, we've gained a lot of experience and no more about the materials, the contaminants that limit the life of the vessel. I think AREVA's addressed that in its current designs.

So we've got analysis, conservative analysis using worse case, nickel, copper content in welds, to show a 60 year life. I fully expect we can justify a longer life, and we'll have more margin in the first generation of reactors.

We haven't done a detailed study of what that life could be, but we do intend to do that.

CHAIRMAN POWERS: It's peculiar when you're planning any project. I've done this once in my life, where you're planning a project that's going to exceed your own lifetime. So it's difficult to anticipate everything that's going to happen in that period of time. But it's interesting.

The other question is do we have the empirical data you need for doing those analysis out there, you know, into the beyond 60 year? Now this has a fairly low fluence onto the vessel. So the empirical database may be adequate for you.

MR. FINLEY: And at this point, I'm not aware that it's adequate. But certainly we'll have the operation between now and whenever we choose --

CHAIRMAN POWERS: You've got a while.

MR. FINLEY: We'll have the benchmarking done. We'll have the samples and tests done that would support, you know, knowledge of what the brittle fracture situation is on the vessel and we would extend accordingly.

CHAIRMAN POWERS: One of the challenges, of course, is by the time you get to the point you want to think about extending the license even to 60 years, you really don't know what the technology that you're going to have an end will be.

I mean 20 years is a lifetime in technical evolutions nowadays, and 40 years is beyond anybody's planning horizon for technical developments. So doing too much now is kind of a waste of time.

MR. FINLEY: We want to keep the option open, but we're not going to, at this point, put a lot of technical resources into justifying an extended --

CHAIRMAN POWERS: It's just thinking and aspirations, and we're really fishing for it. It's an interesting question, and it's kind of fun. It's great. Well, oh. Are there any other questions on this issue?

(No response.)

CHAIRMAN POWERS: Go ahead.

MR. CARNEAL: Okay. I'll ask Shanlai Lu and Fred Forsaty to approach.

(Off the record comments.)

MR. CARNEAL: Okay, good morning. My name is Jason Carneal. I received a BS and MS in Engineering Mechanics from Virginia Tech. Subsequently, I went to work at the Naval Surface Warfare Center at Carderock Division for four years as a mechanical engineer, where I performed experimental studies on Naval hydrodynamics.

I came to the NRC in November of 2008, and since then I've served as Chapter of PM for Chapters 4, 6 and 15, in the EPR Design Center.

Today, this morning we'll be presenting the staff's evaluation of Chapter 4 of the Calvert Cliffs nuclear power plant Unit 3 combined license application.

The technical staff that were involved in this review included members from the Reactor Systems, Nuclear Performance and Code Review branch, and the Component Integrity Branch. Again, the project managers are myself, Jason Carneal, and the lead PM is Surinder Arora.

Chapter 4 contains information on fuel

system design, nuclear design, thermohydraulic design, reactor materials and functional design of reactivity control systems. As the applicant noted, there is no site-specific information contained in the Calvert Cliffs application. However, the staff does have two open items relating to this chapter.

One open item in Section 4.2, Fuel Systems Design, and one open item in Section 4.3, Nuclear Design. The open items that we've identified include tracking the status of the exemption request for the use of M5 material that is currently identified in Part 7 of the Calvert Cliffs nuclear power plant application.

The staff has requested that the applicant add a discussion of this exemption request to Chapter 4 of their FSAR.

The other open item, RAI 225, Question 4.3-1, tracks the need for the applicant to address the need for a COL information item on fluence methodology benchmarking that was identified in the safety review of the U.S. EPR design certification application.

To discuss the details of these open items, I'm going to turn the presentation over to Mr. Fred Forsaty.

MR. FORSATY: Good morning. My name is Fred Forsaty. I'm an engineer at the new reactor with Calvert Cliffs. I've been here for about four and a half years and in about six months I'll be eligible to retire.

(Laughter.)

MR. FORSATY: I have a Bachelor's in Chemical Engineering from Madison, Wisconsin. A nice place to go to school. A Master's from the same place in nuclear. Then I moved to Penn State University. I spent about six, seven years doing fuel management and electronic type of work.

After I got out of school, I built two simulators for Arizona Power and the Perry plant. Did the core and thermohydraulic part of it. I had also experience working at Yankee Atomic, Beaver Valley, Niagara Mohawk, Perry Plant, V.C. Cook and worked with the Swedish on the Oskarsham plant and have been a consultant for Westinghouse, GE and B&W.

Having said that, I'm going to start the presentation on Section 4.2, on the fuel design, fuel system design. This section of the COLA incorporates by reference the U.S. EPR design certification application. We have a RAI that tracks the ongoing review of the U.S. EPR FSAR as an open item.

On the M5 exemption request, the COLA for the Calvert Cliffs includes an exemption on the use of M5 rod cladding material. We have done an evaluation on this item, and we have concluded that the FSAR did not contain a reference to the exemption request for M5 material, and the staff is currently reviewing the exemption request.

Open items are RAI 226, which requests a discussion to be added to Chapter 4 of the Calvert Cliffs COLA FSAR and review used to track the status of the exemption request.

CHAIRMAN POWERS: I understand why the exemption request is there. It's because of the way the regulation is written. When you review the exemption request, what are you looking for?

MR. FORSATY: Well, that's a good question. Basically initially, I would look at, to ensure that the changes in the cladding material, M5, and use of that would not increase the consequences of any accident, such as LOCA or rod ejection or anything related to that. That would be my starting point. Then I would start looking for the other things.

Actually, we have ten, or we have started this type of review anyway, so we've got a very good idea what the impact would be, for example, on large-break LOCA, or if there is an impact on the small-break LOCA.

But it's still that of course, we are in the review process right now. We have not made a definite conclusion or decision, but we are getting very close.

MR. LU: And it's similar to what are, we have been doing with for operating fleet. So we have granted many exemptions by our guys. It's a similar process actually.

CHAIRMAN POWERS: They've just basically seen to it that it's about like those that you've already granted. So it's not a big deal here. Do you suffer from any lack of empirical data in these reviews? I mean it's not a question I really expect an answer to. But if you have an answer, I would take it.

MR. LU: I think the empirical data is always helpful. The more, the better. But as part of the topical report review on the fuel, I think that Fred is the lead and we are in the process to review that.

MR. FORSATY: I think you're asking good question. I think what's lacking right now is that enough data in the high burn-up area. But then we have enough to, you know, get a good review of that.

CHAIRMAN POWERS: When you think about burn-up, do you think about them getting up to the limit of, say, 50-60 gigawatt days per ton, or do you think any beyond that?

MR. FORSATY: Right now, the approved rod limit is 62 gigawatt day and a rod limit of 62, would translate to about 56 to 57 assembly average. That's what basically you're looking at. 56 would give you three cycles. But I think in future, the applicants, some of the applicants are looking to go beyond 70 for rod.

CHAIRMAN POWERS: Yes.

MR. FORSATY: So it's good to have.

(Laughter.)

CHAIRMAN POWERS: Okay. Well, that was kind of an aside.

MR. FORSATY: But yes. There are not that many plants that go to four cycles, at least not in U.S. I haven't seen any. Overseas, I have seen, you know, Europeans. There are some plants that have gone to four cycle. So going beyond 62 for a rod max burn-up, I don't know what the benefits would be at this point. Or maybe I don't understand.

CHAIRMAN POWERS: Yes. Most cores, it's a dead loss for you to go to four cycles. I mean nobody's doing it. That doesn't mean that's going to be the case in the future.

MR. LU: Next one.

MR. FORSATY: Okay, moving to the next topic, that's the new section. For the Section 4.3 of the COLA, FSAR incorporates by reference the U.S. EPR design certification application. Again here, we have a RAI that tracks the ongoing review of the APR FSAR as an open item.

The fluence methodology benchmarking in U.S. EPR RAI No. 344, AREVA, we have asked actually, requested from AREVA to provide a COL item in the FSAR for the U.S. EPR, to address plant-specific surveillance of the reactor vessel in regard to fluence methodology or fluence methodology benchmarking.

In a letter dated April 2010, UniStar stated that when the COL item is added to the U.S. EPR FSAR, the applicable parts of that COL application for the Calvert Cliffs plant Unit 3 would be updated to address this additional requirement.

We did our evaluation on the COL FSAR. We'll need to address this COL information item once it is added to the U.S. EPR FSAR. Then we are tracking this open item, RAI 225, which requests an update for the COL FSAR Tier 2 to include the additional requirements of that COL item.

We have discussed this fluence benchmarking previously as a part of the ACRS presentation for 4.2.

CHAIRMAN POWERS: I guess the question that comes up on this is the same we asked before, you know. The plant's designed for 60 years. You're going to give them a license for four; they're going to get like standard for another 20. But it's likely to be good beyond that. But what you do think about life beyond 60?

MR. FORSATY: 60 of the plant or 60 of us?

(Laughter.)

CHAIRMAN POWERS: We don't like to think about beyond 60 for us. I don't have any fluence benchmarking on myself.

MR. FORSATY: Shanlai is an expert in aging, so I'm going to --

(Laughter.)

MR. LU: I think that one of the unique feature of the EPR is it uses the hydrogen flector, to reduce the fluence level on the vessel wall. Actually so comparing with operating fleet, it is much better situation, that's number one.

I think the second is if you read the RSE and FSAR related to the fluence calculation. They have already done up to 60 full power operational years. Not just 60 years. In the 60 years, you always have, you know, one half year, two years out of the cycle. You have a couple of months of, you know, the outage.

So really I think right now the numbers provided by AREVA has reached the point that I think all we need is just to verify the methodology as part of surveillance, to ensure that the calculation methodology is conservative. From our perspective, the vessel fluence is low.

MR. FORSATY: We also had a couple of experts that helped us to do this analysis, that have done previous work in this area, and they are retired NRC staffers.

CHAIRMAN POWERS: Yes. But again you have, you have the same problem that the applicant has, is that you have no idea what the technology's going to be. By the time you get around to these, even the first license extension you'll know, and to go beyond 60, I mean, my crystal ball is very, very cloudy for that time period. But you assume that it will be much better technology at that point.

So the question I pose to you is do we lack, do we foresee any lack of adequate empirical databases here, and again, we come back to your point, that there's a shielding on this. So there's a relatively low fluence on the vessel. So maybe the adequate -- maybe we don't have an inadequacy in the empirical database, as far as we know right now.

We could discover that there is some strange thing in the water that classes something that we've never seen before, either good or bad in the future.

MEMBER SHACK: You've changed the spectrum of the shield.

CHAIRMAN POWERS: Yeah, I changed the spectrum, things like -- I mean all kinds of things could happen. I think --

MR. FORSATY: If you change the fuel management scheme, that could have an impact if you go from a low leakage to a high leakage, or if you use your fast neutrons less often or more often, that could impact your fuel.

That is not going to be substantial, I think. I think we're pretty much convinced that with the experience that we have had previously on the NRR side, we are pretty much convinced that our decision to go forward with this, we are very comfortable with that.

MR. LU: But we do require surveillance.

MR. FORSATY: That's right.

MR. LU: That's the -- they need it pick up the capsule and a measure of that, to benchmark the measure --

CHAIRMAN POWERS: Yeah, and right now we're running into lack of capsules for the license extensions and things like that. I mean I just presume that we'd just as soon not do that in the future.

MR. FORSATY: There would be some consideration at some point. Any more questions? That's my favorite part, conclusions. The staff is --

(Laughter.)

MEMBER SHACK: Oh, come on. You just love talking to us.

MR. FORSATY: My boss always tell me don't forget to be brief. The staff review confirms that the COL applicant get required information relating to the reactor within the exception, or with the exception of the identified open items, and COL applicant is expected to get the outstanding information in the COL FSAR related to this chapter. That concludes our presentation. If you don't have any other questions.

CHAIRMAN POWERS: Fred is desperate for questions. Does the committee have more questions they'd like to pose here? He's going to feel very lost if we don't interrogate him closely on these things.

Well, I do encourage you whenever you find things, where it would be useful to have additional empirical databases or computation tools in your review that we flag those things. Though that won't be part of this discussion, I mean part of anything we write on this, they will be on other things.

It's useful to the ACRS as a whole to know those things. Gee, it would be nice if we had this, or I can anticipate a need that will -- there will be in the future of needing these things.

Those may be some of the most useful things to come out of that, you know, kind of the spinoffs, as they say in the NASA programs that come from these reviews, that would be helpful for us to know about.

So if things come to mind, don't hesitate to pass them on to us, because Fred may not be here for the license extensions, but somebody's going to be. Since we don't have any reason to dislike that guy right now, we might make his life a little easier. That's right, suffer just like I did. Thank you very much.

MR. FORSATY: Thank you.

CHAIRMAN POWERS: We're in the familiar position of being way ahead of schedule, which is good.

(Off the record comments.)

MEMBER SHACK: This is the Subcommittee, so we can go on just plowing ahead.

CHAIRMAN POWERS: Yes. Yeah, we'll just plow forward.

(Off the record comments.)

MEMBER SHACK: Well, the only problem will be our public commenter if he doesn't show up on time, or if he doesn't up early. Or if he doesn't show up early, then we're going to wait.

CHAIRMAN POWERS: For who Mike?

MEMBER SHACK: No, Jim. Jim August.

MEMBER STETKAR: Oh, I asked him to come early.

MEMBER SHACK: Oh you did?

MEMBER STETKAR: He should be here by

2:00.

CHAIRMAN POWERS: Does he want to make his comment then?

MEMBER STETKAR: I had him on schedule to be last, so he --

CHAIRMAN POWERS: That's kind of traditional being last, but there's nothing --

MEMBER STETKAR: No. He's coming early so that he's prepared in the event we move quickly. He's going to watch the Chapter 17 discussion.

CHAIRMAN POWERS: Okay.

MEMBER STETKAR: I have a question.

CHAIRMAN POWERS: We didn't drive you away the first time. You keep coming back.

MR. GIBSON: I heard there can be some licensing work for an extension. I ought to be prepping for it now.

CHAIRMAN POWERS: Yes.

MR. GIBSON: I will look forward to that day.

CHAIRMAN POWERS: Yeah. Well, I don't think you're going to actually do it, Greg.

(Laughter.)

MR. GIBSON: If I can live that long, I'll come back and do it free.

CHAIRMAN POWERS: Careful what you wish you.

(Laughter.)

CHAIRMAN POWERS: Chapter 5.

Chapter 5, Reactor Coolant and Connector Systems

MR. GIBSON: Chapter 5. We've already talked about our methodology for approaching that, so what I'd like to do is go to the next slide please. With regard to Chapter 5, we'll be presenting it. We did not have, as to Chapter 4, any departures from the certified design. We had no ASLB contentions.

Here, we had six COL information items that Mark will be presenting, and three NRC open items and two NRC confirmatory items. So we did have a little more meat on this presentation. Mark Finley will be joined by Dale Matthews, and I'll turn this over to you now Mike.

MR. FINLEY: Okay. Thanks, Greg. Good morning again. As Greg said, I'll be supported by Dale Matthews from AREVA. I'm on Slide 5. The focus of the presentation is on the site-specific information that supplements the U.S. EPR FSAR.

Slide 6. The organization for the Chapter 5 discussion is COL information items, SER open items, SER confirmatory items and then conclusions, back to

Greg.

Slide 7. To start with the COL information items. The first item has to do with identifying any additional ASME code cases that we might take advantage of, and essentially we don't plan to take advantage of any additional ASME code cases beyond what AREVA has specified in the design cert at this time.

Slide 8. The next COL item has to do with identifying the implementation milestones and the applicable addition of the code that we'll use for the pre-service and the in-service inspection programs. Regarding the pre-service inspection program, we'll have that program implemented prior to the initial start-up.

Regarding the in-service inspection program, that will be implemented prior to commercial service. Small difference in time frames there, but that's the milestone that we have. The pre-service and the ISI programs for the red cone pressure boundary will meet the requirements of 10 C.F.R. 55 Alpha, and comply with ASME boiler and pressure vessel codes Section 11 2004 addition. So that's the addition of the code that we will comply with.

CHAIRMAN POWERS: When you think about

issues like pre-service inspection, I had no idea how many issues that we have up here before the ACRS, unless something has happened with material and they go back and look at the previous inspections and well, we didn't look at that sort of thing and stuff like that.

So the design of a pre-service inspection system is not an easy thing, is it?

MR. FINLEY: No, it's not and we've incorporated some of that experience that you talk about in the design, the geometry, the materials that we're using. Dale, you might be able to speak more about how we used operating experience.

MR. MATTHEWS: My name is Dale Matthews.

MEMBER STETKAR: Speak into the mic.

MR. MATTHEWS: My name is Dale Matthews. I'm supervisor of Component Design for AREVA for U.S. EPR. We've faced on some difficulties. I've got a little bit of background in field work, and based on some difficulties we've had doing in-service inspections on the firs generation plants, we have done extensive review of all the component designs, with practicing Level 3's who current do in-service inspections, to make sure that all the required inspections can be performed using current technology.

CHAIRMAN POWERS: You put that codicil in that says "current technology," and of course you can't do anything but current technology. But quite frankly, our ability to inspect metal components for the things that we're interested in, which are things in the metal, is really bad, you know.

I mean I keep having this vision of, you know, when the starship Enterprise, you know, from hundreds of miles away. They say "Well, our sensors detect a four on the shielding on this spaceship that's located hundreds of thousands of miles away" and what-not, and we can't do that. That's terrible, isn't it? I mean it really is? It's something that we ought to do better.

MEMBER SHACK: The upside is at least we can inspect it with current technology. We're not waiting for the guy to develop his super X-ray vision, so that we can finally look at the vessels.

CHAIRMAN POWERS: That's true, that's true. But the metallurgists have just not -- they are not up to Spock's standards here are they?

MEMBER SHACK: No.

CHAIRMAN POWERS: It would be interesting to note, but it would be useful.

MEMBER SHACK: This language always

confuses me. "The ISI program will conform with the 2004," and of course the very next bullet says "The ISI program will be the 12 months before." I'm pretty sure you're not going to build this fast enough. It's going to be 12 months from 2004.

MR. FINLEY: Certainly for the first inspection integral, we would plan to incorporate the 2004 edition for future inspection integrals then, whatever the addition.

MEMBER SHACK: So this would be the first --

MR. FINLEY: 12 months prior to the successive 10-year inspection integrals. Then we would update the requirement.

MEMBER SHACK: Oh, that's the trick, isn't it? It's a ten-year inspection integrals. So you --

MR. MATTHEWS: Every ten years, you would update your addition --

MEMBER SHACK: You'd update your addition.

MR. FINLEY: Yes, and that's the second bullet here, that for successive 120 month inspection integrals, we'll comply with the latest addition and addenda, the code approved in 10 C.F.R. 5055, 12 months before the start of that integral.

Slide 10. We don't have any code relief

requests that we intend to use, but if we do apply for a relief request, it will of course be through the proper process with the appropriate justification. But right now we don't have any relief requests in mind.

Slide 11, which is the next COL information item regarding the material surveillance program. It's a schedule COL item. When will we implement this program, such that we're committed to implementing the reactor vessel material inspection program or surveillance program prior to the initial fuel load.

Next item is Item 5.3-2 is regarding the plant-specific pressure and temperature limits report, PTLR. We need to provide that report. We haven't provided it yet, and we need it confirm that it was in accordance with the generic and approved methodology.

We will provide this plant-specific PTLR for Calvert Cliffs Unit 3, for technical specification 5.6.4. It will be based on the AREVA methodology that's approved, ANP-10283P.

Slide 13. Next item, COL information item had to do with providing the plant-specific reference temperature for pressurized thermal shock, in accordance with 10 C.F.R. 5061 for belt line materials.

Essentially, when we confirm the materials that we have our vessel through the fabrication process, we will provide this reference temperature. We'll provide it to NRC staff within one year of acceptance of the reactor vessel on site.

Slide 14. This regards the steam generator inspection program, to identify the addition and addenda to Section 11 applicable to this site-specific steam generator inspection program.

Essentially, the steam generator program tube inspections for pre-service inspection and the initial ISI integral will comply with ASME boiler and pressure vessel code, Section 11, 2004 edition. Again, no relief requests or alternatives are required for use of this 2004 edition.

CHAIRMAN POWERS: Correct me if I'm wrong, but I think when we asked AREVA for the design certification, what was the highest worker dose activity at the plant. It was in fact the steam generator inspection program that came up, that they identified.

Now they use a fairly conservative methodology in making that identification. But of all the activities, this inspection was the number one on the hit parade for them. You guys give an additional thought to this inspection program, to try to reduce worker dose.

MR. FINLEY: Certainly, we have. In fact, in reviewing the specification for the steam generator, there's processes that you can use on fabrication of the bowl, including the smoothness of the surface and/or use of polishing techniques, and we looked at that very strongly to reduce contamination and dose for this evolution.

As you say, it's one of the chief contributors to personnel dose. So that's been incorporated in our steam generator specification review process.

CHAIRMAN POWERS: Good.

MR. FINLEY: And it's based on industry experience of plants who have gotten replacement steam generators and used those techniques. So we're ceding that fact in the design of the new components.

CHAIRMAN POWERS: Okay, good.

MR. FINLEY: Also regarding future 12 or 10 year inspection integrals for the steam generator program, we will incorporate the latest addition and addenda to the boiler and pressure vessel code approved in 10 C.F.R. 5055 Alpha on the date 12 months before that -- I'm sorry. Wrong bullet.

But for the initial fuel load, we will incorporate the addition in an addenda 12 months before the initial period. However, we intend that to be a 2004 edition at this time.

Slide 15. For successive ten-year periods or a 120 month period, we will incorporate the latest edition and addenda approved by 10 C.F.R. 50.55 Alpha 12 months before the start of that integral. So similar to the reactor vessel.

Again, we have no relief request that we expect to apply to this inspection program. But should they be required, we will provide the appropriate justifications and go through the NRC review and approval process for that. Any questions on the COL information items?

(No response.)

MR. FINLEY: Okay. Slide 16, just to say we're moving next on to the SER open items, and there are three of those. If you turn to Slide 17, and the first two have to do with RCS leakage and procedures and information that will be developed as we develop operating procedures to manage RCS leakage.

RAI 223 requests procedures for conversion and alarm set points for prolonged unidentified leakage inside containment. We have committed in the letter that you see here to provide those procedures.

They haven't yet been developed to take advantage of essentially three indications that we have that can fairly accurately determine low level leakage, gaseous and containment activities in a containment atmosphere, containment sump leakage rate and containment cooler condensate flow. These are methods that you can get to one GPM or below accuracy.

So we intend to use those indications and develop tools for the operators to apply to the indicated levels in the control room, to calculate the leak rates.

The third item, RAI 227, has to do with FSAR Table 13.4-1. We had a reference to Section 5.4.2.5 under the pre-service testing program, and we need to move that to the pre-service inspection program, and we will do that.

That was it for the open items. Next Slide 18, onto to the SER confirmatory items, and Slide 19, essentially two confirmatory items. I mentioned one already, the plant-specific PTLR. We confirm, we commit to providing that prior to the initial fuel load, and we'll incorporate that into FSAR Section 5.3.2.1 and Part 10 under ITAAC.

Secondly, in response to RAI 40, we will update the same table I just mentioned, that 13.4-1 table, to include a reference to Section 5.4.2.5, and this is in the in-service inspection program section of that table. So an administrative item, but we'll include that reference.

Then we have a Revision 7 planned to the COL log Greg. Is there a roughly time frame?

MR. GIBSON: September 30th.

MR. FINLEY: September to incorporate these changes to the COLA.

CHAIRMAN POWERS: That's just a -- to be clear, that is just your best guess right now for the --

MR. GIBSON: That's correct. It depends on the finalization of some of the design certs, because that's the principal reason for Rev. 7.

MEMBER RAY: Would you go back to 17 please?

MR. FINLEY: Yes.

MEMBER RAY: What is the -- number two. I didn't follow. I missed it.

MR. FINLEY: Okay.

MEMBER RAY: I focused on --

MR. FINLEY: Yeah, I might have skipped

it, so thank you.

MEMBER RAY: I think you did.

MR. FINLEY: So Item No. 2 is RAI 223, and it's related to the procedures to measure and respond to unidentified leakage inside the containment. This relates to specifying operator actions to take, if we see any increased RCS leakage.

MEMBER RAY: I can read it. What is your response?

MR. FINLEY: So we haven't yet developed these procedures, but our response is that we will develop these procedures prior to start-up, and have them in place. So we'll identify the operator actions. We'll incorporate those in procedures. We'll have them available prior to start-up.

MEMBER RAY: What is this triggered by? I mean prolonged low level leakage forecast?

MR. FINLEY: I think this terminology comes from the Davis-Besse boric acid leakage. If have low levels of unidentified leakage for long periods, you can cause other problems. So --

MEMBER RAY: You had listed "or cited," your ability to detect by signals from the gas phase, the sump and a third item, which I didn't catch.

MR. FINLEY: Condensate cooler.

MEMBER RAY: Condensate cooler.

MR. FINLEY: Excuse me, containment cooler condensate flow rate.

MEMBER RAY: Well, prolonged low level leakage above normal leakage rates. I guess you've -- this is something that is -- people are addressing or is this new? This is the first time I've encountered this notion of normal leakage and prolonged low level leakage above normal leakage.

MR. FINLEY: No, I think this is prolonged low level leakage.

MEMBER RAY: Period.

MR. FINLEY: Period.

MEMBER RAY: Well, the words say "above normal leakage."

(Simultaneous discussion.)

MR. FINLEY: You're going to --, but I mean right.

MEMBER RAY: It's that increase?

MR. FINLEY: Right. It's that increase that's --

MEMBER RAY: So I'm mulling it over an eight -- you know what to do, I gather.

MR. FINLEY: Yes. We frankly, I'm not aware of any new regulatory requirements in this arena, but of course all operating plants have RCS leakage monitoring programs now.

MEMBER RAY: Yeah. No, I understand that. I'm just trying to figure out how you identify low level leakage above normal leakage. You can do that?

MR. FINLEY: Through those indications I mentioned. Those three indications provide the most accurate --

MEMBER RAY: Okay, I don't -- appreciate it.

MEMBER SHACK: You use slope.

MEMBER RAY: Huh?

MEMBER SHACK: I think it's slope. You can --

MEMBER RAY: Well, I know.

MEMBER SHACK: It's still below.

MEMBER RAY: The implication, Bill, is that there's something that recognizes normal leakage, then there's low level leakage above that. So you've got some way of monitoring that and you trigger when this condition exists. I just hadn't run into that before, and everybody seems to know how to answer the question, so I'll guess we'll -- it's okay.

MR. FINLEY: Okay. I think we covered the confirmatory items on Slide 19, which brings us back to conclusions, and Greg.

MR. GIBSON: Yes. Again, we made a submittal on the 14th which completed numerous items. We know the staff has another opportunity to fully vet those yet. But again, where we are is we took no departures for Chapter 5. We have no ASLB contentions. The three SER open items we have made the submittal, which went to the specific request.

The NRC sent us an RAI. We believe those will be closed. We have two confirmatory items and we will incorporate all these results into Rev. 7 of the RCOLA in the end of the third quarter, beginning of the fourth quarter.

CHAIRMAN POWERS: With the exception of we don't understand everything yet --

MEMBER RAY: I'm still puzzled by how they do that, but I'm willing to be educated. So I don't mean -- the only other comment I'd say, Dana, is that I don't know how you'd answer most of those questions, other than the way they did answer them. They seem to me to be -- there isn't any answer to give, other than what they gave.

CHAIRMAN POWERS: I think in many respects, I believe we'll talk to the staff in a second. But in any respect, you have to understand the FSAR is going to be a living document, and things will change, and you need -- you want a section in there specifically addressing those things that we change, once the plant gets up and running. I think it's like that. So --

MR. COLACCINO: Dr. Powers?

CHAIRMAN POWERS: There's very little to say right now.

MR. COLACCINO: Dr. Powers, this is Joe Colaccino. Maybe I can help here a little bit. If it relates to the confirmatory items, I will just note that the staff doesn't usually bring confirmatory items forward to the committee, unless they present technical topics of interest.

The reason for that is the staff has made a conclusion on those, and they found them acceptable. It's just waiting for the FSAR to be updated.

So when it regards a confirmatory item that's discussed, in this case by UniStar, the staff has already made an acceptable determination on that. So there isn't any additional questions that the staff has. Just looking to see that the item has been incorporated correctly in FSAR.

So that may be giving some confusion with respect to confirmatory items discussed in this forum, because the staff's done with that.

CHAIRMAN POWERS: I don't think it hurts to mention confirmatory items to us. I think we understand what they are.

MR. COLACCINO: I agree, and I just wanted you to understand that if that was causing any confusion, the staff --

CHAIRMAN POWERS: Yes, that did.

MR. COLACCINO: Because that was the comment I was picking up on, is I don't know how you'd answer them any other way -- how they can be answered in any other way than they did.

CHAIRMAN POWERS: I think that's what we say about any of the answers.

(Simultaneous discussion.)

MR. COLACCINO: --about the confirmatory items. We certainly agree with that in this respect.

CHAIRMAN POWERS: I mean some of the answers are just yes, we'll do this and I mean there's nothing else to do except put it in a reference or a statement or a connection and that's fine.

MR. COLACCINO: That is correct.

CHAIRMAN POWERS: And the confirmatory items and what-not give us some comfort that things are actually happening here, and the knee bone is indeed connected to the thigh bone here, and that's what we're after. So maybe it's not the most exciting thing in the world, but that's what we have to do.

MEMBER SHACK: You did make the statements that the PTL limits will be computed by essentially the report. But the last time we looked at this just to review it, that report wasn't. So it's not an approved methodology, unless there's some update since the last time. The SER we have, that is an open item yet.

MF Okay, and I'm not sure of the revision of this methodology here, but we intend to incorporate whatever the most recent approved --

MEMBER SHACK: Whatever it takes to get it approved.

MR. FINLEY: --and reviewed methodology that's available from AREVA.

CHAIRMAN POWERS: Any other questions for these speakers?

(No response.)

CHAIRMAN POWERS: We're getting substantially ahead of schedule here. I'm going to have to chastise Greg for speaking too fast or something. So if we're running into problems with your staff, let me know. But otherwise I suggest that we just press right on ahead. Sorry we're belaboring these issues so much, stretching out the presentations.

(Off the record comments.)

CHAIRMAN POWERS: Surinder, would it be appropriate if we can just go ahead and take a break, and you guys can set it up? Why don't we do that? Why don't we take a 15 minute break and they can -- and not be under pressure.

MEMBER SHACK: We can relax. I mean we're not under a huge amount of time pressure. In fact, I've got --.

(Whereupon, a short recess was taken.)

CHAIRMAN POWERS: Why don't we get back into session?

MR. ARORA: Good morning, again. This is staff's turn to --

CHAIRMAN POWERS: Gentlemen. We're starting now. They're trying to sort out other issues. We'll get back to you. Go ahead, Surinder.

MR. ARORA: So we are going to get back to staff's presentation on Chapter 5, and let me introduce the Chapter PM, Tarun Roy, who will be leading staff presentation, with the help of technical people who also are involved with the application review. Tarun?

MR. ROY: Okay. I'm Tarun Roy. I'm the IBR project manager of Chapter 5, and we have several technical reviewers for Chapter 5. Those who are going to be presenting is John Wu, Tim Steingass, Chang Li, Joel Jenkins, Steven Downey and Greg Makar.

We have -- the staff issued a total of -- okay, we'll go to the next slides. The staff issued a total of 22 questions to the applicant, requesting additional information. Out of 22 questions there are three open items identified in this SER. The staff will discuss these open item in detail. Next.

We have three RAI and it will be presented by the technical reviewer one after another. We have Chapter 5 -- several sections is incorporated by --

(Off record comments.)

MR. ROY: Chapter 5 of the COLA FSAR incorporates by reference the U.S. EPR design -- I'm sorry, the U.S. EPR design certification application, which is currently being reviewed under Docket No. 52-020. Staff, the application stated that this section of the COL FSAR were incorporated by reference, IBR.

The staff reviewed the appropriateness of this information and found it to be acceptable. There are four sections. 5.22, 5.23, 5.411, 5.47, 5.411, 5.412, 5.413 and 5.414 are IBR sections. John Wu will jam in what his recommendation for I21. John?

MR. WU: Okay. Good morning. My name is John Wu. I am from Engineering Mechanics Branch. I'm also the primary reviewer for Section 5.2.1.1 for COL and the standards, and the 5.2.1.2, which is for applicable ASME code cases for Calvert Cliffs Unit 3 COL FSAR.

Section 5.2.1 incorporates by references U.S. EPR codes in the standards in complying with 10 C.F.R. 5055(a), with no departures and no supplements. In the FSAR, there's a COL information items, 5.2-2, which states "The COL applicant should identify additional ASME code cases in its COL application for NRC's review and approval."

So we reviewed, as a result of our reviews, we find FSARs contain additional ASME code cases, and that the applicant indicated there are no additional ASME code cases are planned to be used for the COL application at the time.

So as a result, we have no open or confirmatory items in the SER for this section.

CHAIRMAN POWERS: I guess the question that comes to mind, and may be a fairly benign answer to this, is why did this request go out to them? I mean what motivated you to ask for additional code cases --

(Simultaneous discussion.)

MR. WU: Okay, because the COLs information items were attached to Section 2-2. They say they should identify some of this. So we will ask, you know, yeah, what --

CHAIRMAN POWERS: It's just required --

(Simultaneous discussion.)

MR. WU: Required by the --

CHAIRMAN POWERS: Again, they said we don't have any. That's why we didn't identify it. I understand.

MR. WU: Okay, okay.

CHAIRMAN POWERS: I kind of suspected that was the case, but --

MR. WU: Yeah.

CHAIRMAN POWERS: Okay, fine. Thank you.

MR. WU: Okay, good.

MR. ROY: Next we go to 5.2.1. I'm sorry, 5.2.4. Tim Steingass.

MR. STEINGASS: Back one. There you go. My name is Tim Steingass. I'm the lead technical reviewer for in-service inspection and reactor pressure vessel boundary, reactor coolant pressure boundary. I work in Component Integrity Branch 2.

Under Section 5.2.4, it addresses the pre-service and in-service inspection and testing of Class 1 components and piping. The COL FSAR Section 5.2.4 incorporates by reference the U.S. EPR FSAR SER to Section 5.2.4, with no departures.

However, there is an information item under 5.2.3, which requires that the applicant identify the code of record 12 months prior to fuel load, any relief requests, and the milestone schedule for building of the plant.

The COL applicant stated that they will identify the code of record 12 months prior to fuel load, and the construction milestones in accordance with Table 13.4-1. Now there was some discussion earlier. The code of record for the design of the plant in the PSI program is the 2004 edition of the ASME code.

However, 50.55(a) requires that 12 months prior to the fuel load, the year of the code that's endorsed under 50.55(a) will be used as the basis for the ISI program. So therefore, there may be some code or some relief requests involved with the new year and addenda to the code that comes into effect just prior to fuel load.

In accordance with the requirements of 50.55(a), the applicant stated that they would provide any input for the relief requests in accordance with 50.55(a)'s requirements. And finally, there are no open items. Any questions?

CHAIRMAN POWERS: I thought I understood this perfectly, and now I'm a little confused. They've indicated their design plan according to the 2004?

MR. STEINGASS: Correct.

CHAIRMAN POWERS: And then 12 months prior to fuel load they're just going to tell you that again?

MR. STEINGASS: Well, here's the deal. This may take ten years.

CHAIRMAN POWERS: That's right.

MR. STEINGASS: So it's frankly, it's a good way to go. It's the best way to go, because lessons learned as the code evolves over those ten years are incorporated in the construction, unless they can't confirm with it. Then they would put in a relief request and provide sufficient justification why the plant is safe to operate.

CHAIRMAN POWERS: Yes, and kind of business as usual as a matter of fact.

MR. STEINGASS: Yes sir. Thank you.

MR. ROY: Next will be 5.2.5, Chang Lee.

MR. LI: My name is Chang Lee. I'm from -- Plant System Branch. I'm the lead reviewer of Section 5.25, reactor coolant pressure boundary leakage detection, for both Calvert Cliffs COL application and EPR design certification.

We reviewed the reactor coolant pressure boundary leakage detection with respect to meeting Regulatory Guide 1.45. Another regulatory position is C.3. This position has to do with the procedures that were derived from the lessons learned for operating experience of daily special events.

We were not able to find sufficient information in the design certification application or in the COL application to address this regulatory position. There was no COL information item in the design certification that would require COL to address this issue. Calvert Cliffs FSAR uses the process of IBR without any supplemental information.

In our review, we first asked RAIs in the design certification review, that asking them to identify the COL information item. AREVA indicated in the RAI response that the conformance of this regulatory position relies on the COL application.

So in parallel with the review of design certification, we also asked Calvert Cliffs to provide supplemental information in the FSAR, to address the procedures, the alarm set point in accordance with Reg Guide 1.45 position C.3. So these are open items we identified. So is there any questions?

MEMBER RAY: Well, I asked a question of the applicant, if they knew how to do this. The answer, I guess, is yes, everybody's doing it, so we know how to do it. What does it look like? What is something that complies with the Reg Guide say?

MR. LI: So we're asking them -- currently, we have tech spec requirements at certain limit, 5.5 --

MEMBER RAY: Right. I'm familiar with those.

MR. LI: So that will shut the plant down. But however, in light of lessons learned, Davis-Besse, they are looking, they were looking at like .1 or .2 gallon per minute for many, many months or even years without taking proper actions.

So that was the problems. So we're asking them, as long as they have abnormal leakage being identified, they have to have procedures for operator to docket, to identify to the possible where is leak, and trending and taking the actions to what's practical. We want them to, just because of .1 gallon per minute to shut the plant down. However, whenever they have opportunity, when the plant's shutting down, like refueling stage, they will go aggressively and finding the leakage, the source of leakage and fix it. So that's the whole purpose of this leakage detection procedures.

MEMBER RAY: Okay, fine. So their procedures might have the character that you just indicated they would have. I guess I didn't understand the reference to alarms. I had this vision that somehow you were going to detect long-term low level leakage above normal leakage, which I think were the ones that they used. Those aren't the ones that you used.

MR. LI: Yes.

MEMBER RAY: And I'm just trying to figure out well, what is this? I'm much more familiar with what you're describing now, which is reactor engineers monitor an unidentified leakage over a long period of time, try and figure out when there's some change that they need to --

MR. LI: Right.

MEMBER RAY: Or some amount, whether it's a change or not, that they need to then investigate when they have an opportunity. If that's what you're talking about, I understand.

MR. LI: Yeah.

MEMBER RAY: Okay.

CHAIRMAN POWERS: You're happy?

MEMBER RAY: That I understand, yes. I'm happy that I understand what they're talking, because I wasn't sure what they were talking about before. But now I am. Thank you.

(Laughter.)

CHAIRMAN POWERS: Good. I think we can go on.

MR. ROY: Next will be Joel Jenkins for Section 5.3.1.

MR. JENKINS: Yes. My name is Joel Jenkins. I'm the lead technical reviewer for Section 5.3.1, reactor vessel materials. I'm also going to be presenting for Steve Downey, who can't be here today, for Section 5.3.2 and 5.3.3, and I don't have a card, but --

(Laughter.)

CHAIRMAN POWERS: It doesn't arrive until you get your official 10 card.

(Laughter.)

MEMBER SHACK: Better. This way you can get --

MR. JENKINS: I was hoping that would excuse me. Oh no, just kidding. Let's get started here, Section 5.3.1, reactor vessel materials. This section incorporates the U.S. EPR FSAR, with no departures or supplements, except for subsection 5.3.1.6, which describes the reactor vessel material surveillance program.

In Section 5.3.1.6 of the U.S. EPR FSAR, it states that a COL applicant that references the EPR design will identify implementation milestones for the material surveillance program. Now the COL applicant has done this. They state that the implementation milestones for reactor vessel material surveillance program are provided in Table 13.4-1.

Now that table states that implementation is to be prior to initial fuel load. The staff finds that this COL item is acceptable. It meets the requirements of Appendix H of 10 C.F.R. Part 50. There are no open or confirmatory items in this section.

MR. ROY: Okay. Any question?

CHAIRMAN POWERS: You happy? It's going to be implemented.

MEMBER SHACK: It's going to be implemented, and it's a good program, as far as I can tell.

CHAIRMAN POWERS: It's an adequate program, right?

MEMBER SHACK: Meets the regulations.

CHAIRMAN POWERS: Good. Thank you.

MR. JENKINS: Okay. Let's move on to Section 5.3.2, P-T limits, upper shelf energy and PTS. EPR COL Item 5.3-2 states that a COL applicant that references the U.S. EPR design certification will provide plant-specific pressure and temperature limits using an approved PTLR methodology.

Staff notes that the generic PTLR of the U.S. EPR design was submitted by AREVA. Technical Report ANP-10283(p), Rev. 1, as part of the design certification. In response to an RAI, the applicant confirmed the use of the generic PTLR provided by AREVA, and committed to submit the plant-specific P-T limits prior to fuel load.

The staff found that the applicant's response to the RAI and resolution of the COL item was acceptable, because it meets the requirements of Appendix G to 10 C.F.R. Part 50, and it's consistent with the approach outlined in GL 96-03. There are no open items.

CHAIRMAN POWERS: So will they -- they will submit their pressure and temperature limits to you, based on using the AREVA methodology?

MR. JENKINS: I believe they have committed to submit that.

CHAIRMAN POWERS: And then what do you do?

MR. JENKINS: I think I should probably defer to Neil Ray, the backup technical reviewer.

MR. RAY: Hi. Good morning. This is Neil Ray. Let me address, I think, last or previous question also. What happens in the PTLR area is AREVA submitted a generic PTLR using 96-3 generic letter, and we are currently reviewing it. We have certain several RAI questions and they're responding it.

So the idea here is what Calvert, what these guys, Calvert Cliff folks, they basically commented. They said yes, we are going to use AREVA methodology, and we are going to use AREVA's bounding P-T limits.

However, as we know, the vessel is not manufactured here. We don't know the vessel properties. So AREVA PTLR is based on bounding with the P-T material properties. So when, and when the vessel is manufactured and when they get the actual material properties, at that point they will have done their P-T limits.

And the condition here, they will do it prior to fuel load, which is perfectly acceptable to us, because the P-T limits currently we'll have from AREVA, it will be applicable up to 60 years. So prior to fuel load is about one year or two years, who cares.

CHAIRMAN POWERS: Well I mean the real question is okay, they submit their limits they've used and approved methodology. Now what do you do, discount it or do you go through and look, re-do the calculations, or what is it exactly that you do?

MR. RAY: The idea here is the P-T limits are based on Appendix G, Section 11, 10 C.F.R. 50 Appendix G. Both happen to give Appendix G in this case. The entire methodology will be reviewed and approved by the NRC staff, and currently we are working on it.

So what will happen, when AREVA gets it, the only change they will do, because vessel remains the same vessel; only thing they don't know is the specifically cooper nickel margins, those terms. So when they know those terms, they will basically calculate their adjusted reference temperature, which is an input to P-T limits.

They will, if you're on that program, to develop new P-T limits and -- program, to make sure that the P-T limits are set.

CHAIRMAN POWERS: That's what I wanted to hear. You will do -- you will repeat the calculation?

MR. RAY: Yes.

CHAIRMAN POWERS: And it's your analysis methodology could occur at 0.9 different than theirs, or is it the same as yours?

MR. RAY: No, it is different. It is different. It's completely independent calculation. There is no uniform calculation so that everybody using the same numbers. No.

CHAIRMAN POWERS: Again, the answer that I was looking for. Thank you. Thank you very much. That helps.

MR. ROY: Next slide. Joel?

MR. JENKINS: Section 5.3.3. EPR COL Item 5.3-3 states that a COL applicant that references the U.S. EPR design certification to provide plant-specific reference temperature for pressurized thermal shock values in accordance with 10 C.F.R. 50.61.

To address the COL item, the applicant

provided a license condition in Part 10 of the COLA, which states that the plant-specific RT_{pts} values will be submitted to the NRC within one year of acceptance of the reactor vessel.

The staff found that the applicant's resolution of this COL item was acceptable, because it provides reasonable assurance that the requirements of 10 C.F.R. 50.61 will be met. There are no open items.

MR. ROY: Any questions?

CHAIRMAN POWERS: Again, this is handled in exactly the same way? That is, that once they have the specifics of the vessel and what-not, that staff goes through and verifies the calculation, and then says "thank you." Is that correct?

MR. RAY: Well yes. The big answer is yes. Again, what is happening, just to reconfirm to everybody here, this new vessel materials are much, much better than current vessels that we are using the PWRs.

So the bounding PTS limits which AREVA submitted, which is way lower than the spinning criteria that we are used to in 10 C.F.R. 50.61. And so as Joel said, that when we get the vessel, when they get the vessel material properties, they will update the calculation and we'll look at it and we don't see any problem there with that methodology.

CHAIRMAN POWERS: They do these limits and you just review the material, or do you do an independent calculation?

MR. RAY: No. We do independent calculations. There's a complete independence here.

CHAIRMAN POWERS: That's good. Thank you, Mr. Ray. Surinder, just as a point of information, when we do independent calculations like this, it's useful to know that, and as I said before, it's also useful to know if you think the technology we have available is geriatric or inconvenient or incomplete or anything like that. It's very useful for us to understand that, not so much for this application, but for the broader determination.

I mean in some cases you just review material; in some cases, you do independent calculation. That distinction is useful for us to know.

MR. ROY: Okay. Next, Greg Makar for Section 5.4.2.1 and 5.4.2.2.

MR. MAKAR: Thank you. Well the material in these two sections, steam generator materials and design and the steam generator inspection program, just to review, has all been incorporated by reference, with no departures.

However, there is a COL. There's COL information that the applicant needs to provide, and that is to identify the ASME Code Section 11 edition and addenda that will be applied to the steam generator inspection program. That includes both the pre-service inspection and the in-service inspection.

The applicant has provided that information. It's standard information, which means it's going to apply to all the EPR COL applicants.

What they've stated is that the 2004 edition would be used for pre-service inspection, and the first in-service inspection integral and that the edition that's in our regulations, 10 C.F.R. 50.55(a), will also be -- will be applied to the first in-service inspection integral. That's -- I'll get back to that, because I understand that sounds a little confusing.

And that for the successive integrals, it will be whatever's in 10 C.F.R. 50.55(a), 12 months prior to the beginning of that. So that is the information we did, that we reviewed for this review.

Now for the pre-service inspection, their answer, the 2004 edition conforms with or complies with our NRC regulations in 50.55(a). It also, that is information they have to provide now, 2004 edition, and that's what it's 50.55(a). So that is applicable to the first in-service inspection. However, when that actually -- when that integral actually begins, it could be another edition of the code that's in our regulations.

So that's how I see this as being acceptable, because they provide an answer that's applicable and acceptable today, and is acceptable for when this first integral actually starts. Then of course any successive integrals will also, as they -- with the information they provided is acceptable and complies with our regulations.

Now we did have some questions about the request for additional information about the wording of this information, just to make sure it was consistent with the regulations. We understood what they're saying. They are, they have made some modifications to their Table of Operational Programs. We want these two programs, ISI and PSI for steam generators, to be very visible.

They're not separate operational programs, but we reference these sections in there. One of those changes is, as you've heard, it was put under pre-service testing instead of pre-service inspection. So although that's the open item we have here, we expect that will become a confirmatory item very soon.

CHAIRMAN POWERS: Again, the pre-service inspection figures in an enormous number of incidents that have come before the ACRS, and we continue to learn what we should have inspected for and didn't before. Does that figure in when you look at this kind of material?

MR. MAKAR: Well, there are two kinds of -- I see two kinds of problems that could arise from lack of pre-service inspections. One is failing to adequately inspect for the things that you could already know about, or things that -- but then there may be things that haven't emerged yet.

And personally I'm not the kind of creative out of the box thinker that could anticipate those things for the materials. I think these materials have been in-service together, this stainless steel, carbon steel, alloy 690 material.

They've been around for a while. So I'm not seeing any, you know, new mechanisms. I think we know a lot of the mechanisms and we have the tools to inspect for them now.

So in my view, as these -- as people build these new steam generators, which are very much like the replacements being generated, they know what they're looking for. I think that the bigger, of course, there are real questions, like you've asked before about what if the pre-service inspection was so long ago that your ability to retrieve the data is somehow an issue?

But I think we are, know what mechanisms to look for at this point. I'm not concerned about that for this.

CHAIRMAN POWERS: I mean you're absolutely correct. I'm asking you about what is it about the things we don't know about? I am asking you to be prescient here, and you know, it's a struggle. I admit that. It's why you get the big bucks, to think about these things.

I'm just asking how we're thinking about this, because there is no answer to my question. I mean, you know, what is the stuff that we don't know? It's a mystery.

MR. MAKAR: As you know, each tube will be inspected, go to full length with the bobbin coil, and the more detailed inspections will be done at certain locations.

CHAIRMAN POWERS: Yeah.

MR. MAKAR: And what the DCD says is how you do that is to use the inspection methods that you expect to be used during the in-service inspection integrals. I mean that could be -- there could be something really good that comes along that is helpful later --

CHAIRMAN POWERS: I sure hope so.

MR. MAKAR: --PSI information for it, and that we may not be able to take full advantage of it.

CHAIRMAN POWERS: Yeah. I sure hope our technology for inspection tubes improves. I know I'm putting my hopes on the metallurgists and one should never do that. But --

MR. STEINGASS: I'd like to comment on that issue about what was just discussed. The way Section 11 is constructed is that for pre-service inspection, there are requirements that the data be maintained for the life of the plant.

But also, under the in-service inspection program, it requires that the results of an in-service inspection examination be compared with previous in-service inspection examinations or, if a previous in-service inspection examination doesn't exist, compare those results with the pre-service inspection.

So what you will find then is if there is a new failure mechanism, just by virtue of the fact that there's a difference in the data, you know something is going on. It may not be interpretable. For instance, under ultrasonic examination, the recorded data is compared against previous data, just to see if there's any change in the data.

Then from that point, additional examinations are performed to help interpret what's going on. So if there's any new failure mechanism that should surface, the ASME code has controls in it, built into it, to identify those new surfacing issues.

CHAIRMAN POWERS: Yes, thank you. He makes a sound point. Presumably we will detect something if something new is happening. We may not.

MEMBER SHACK: It wouldn't be the first time.

CHAIRMAN POWERS: Wouldn't be the first time. I mean it's just a very difficult area. Are there any other questions on this particular item?

(No response.)

CHAIRMAN POWERS: Seeing none, thank you.

MR. ROY: Okay, that's it. We are done.

MR. ARORA: That concludes our Chapter 5 presentation.

CHAIRMAN POWERS: Are there any other questions on Chapter 5? What I would like to do is to -- we'll take a short break to set things up, is to proceed on with quality assurance, if we can. I don't know whether all the speakers are here.

MR. COLACCINO: Dr. Powers? Dr. Powers? There's no one from the staff here.

CHAIRMAN POWERS: UniStar can. Let me check with staff. Is that going to pose an imposition?

MR. ARORA: The next one we had was Chapter 12.

MR. COLACCINO: Dr. Powers, we have the staff here to support Chapter 12. We do not have anybody to support 17. I've been trying to contact them. The PRA folks are not, appear not to be available at this time. We can make additional effort. But we do have support to move ahead with Chapter 12.

CHAIRMAN POWERS: Okay. I would just as soon wait until I have my support for Chapter 12.

MR. COLACCINO: I understand. So we're both struggling to support the two chapters.

CHAIRMAN POWERS: But what I would propose we go ahead and do is we'll take a short break so that

UniStar can set up, and we will do their section on 17 and we will defer 17 --

MR. COLACCINO: We'll make additional efforts to get the 17 staff in here.

CHAIRMAN POWERS: I don't know that we need to make heroic efforts. You do what you can.

MR. COLACCINO: You know, I think the other options is we could start that after lunch. I'm looking at Derek here, and maybe that's the better option to go with.

CHAIRMAN POWERS: Yes. Well, we'll do what we can, as we can, and go ahead. Thank you very much for the work on Chapter 5, and we if we can go ahead and start. We'll take a break until ten of.

(Whereupon, a short recess was taken.)

CHAIRMAN POWERS: Let's get back into session. Thank you for letting us move this forward. As I said, my intention is to try to break for lunch around noon, and so if it looks like we're going to go beyond that, just find a logical breaking point, and we can come back. I think even at my advanced age, I can retain thoughts for an hour or two. After that, not so easy. And so I guess Greg?

Chapter 17, Quality Assurance

MR. GIBSON: Yes, thank you. Our

presentation on the FSAR Chapter 17, Quality Assurance and Reliability Assurance and Maintenance Rule, is going to be broken up into three parts. I'll get the first three slides, because they're the same that we've talked about before in the introductions.

Our presentation today, we're going to have Mark Harvey, who is the director of our Quality Assurance and Performance Improvement Group. Rick Szoch, on my right, is the director of Testing and Programs Development. Gene Hughes is our acting director of PRA, and we're supported by our team mates at AREVA, Charles Tally and Josh Reinert. So we appreciate that.

We're going to focus on, as we have, the same format that we've done on our other presentations, and we hope to move smartly through this. At least our dry runs were less than an hour, so I think we have a good chance of finishing before lunch.

CHAIRMAN POWERS: I think not a chance, no.

(Laughter.)

MR. GIBSON: Okay. So with that, I'd like to do Slide 5 and talk about quality assurance, and with that, introduce Mark Harvey.

MR. HARVEY: Okay. Thank you, Greg, and good morning everyone. I had to make a change to my presentation, because I originally had good afternoon. So Change No. 1. Anyway, I'd like --

CHAIRMAN POWERS: We would have written a vicious letter had you not made that change. We would have really said these guys don't even know what time it is.

(Simultaneous discussion.)

MR. HARVEY: --and if I failed to do so, I would write a condition report on it.

(Laughter.)

MR. HARVEY: Okay. I would like to cover the quality assurance portion of the presentation, and the things we're going to cover today are the COL information items, oversight activities taken today, NRC SER open items, NRC SER confirmatory items, similar to the previous approach. Next slide.

A little bit about myself. I have presented before the ACRS previously in my role at General Electric. I have over 29 years of experience in the nuclear industry. I've held various positions in Chemistry, Operations, Radiation Protection, Quality Assurance, training project management, and as now Quality Assurance.

My background includes quality management positions, various quality management positions at New Hampshire Yankee and Florida Power and Light, GE, Hitachi and now at UniStar Nuclear. I started at UniStar Nuclear last September, that's September 2009.

I've held various positions in quality management, including supplier quality manager, audit manager, nuclear oversight manager, quality manager, Vice President of Quality for Nuclear Plant Projects, prior to my current role as Director of Quality and Performance Improvement.

I have extensive experience with nuclear suppliers through my association with the Nuclear Utilities Procurement Issues Committee, what's known as NUPIC, as the Florida Power and Light NUPIC representative and also chairman of the Nuclear Fuels Committee, which is part of the Executive Committee with NUPIC.

I'm also a certified lead auditor with extensive experience in the implementation of QA programs, and I have conducted and been involved in oversight of internal and external audits, supplier audits and surveillances and self-assessment programs. My responsibilities include oversight and management of the UniStar corrective action and self-assessment programs. Next slide.

Okay. Today, my presentation is going to provide a summary of Section 17.1, .2, .3 and .5, and we've talked about several sections being incorporated by reference. That's Section 17.1, Section 17.3.

Section 17.2, Quality Assurance During Operations Phase, we do have one open item associated with that section. But that's really addressed in Section 17.5, which is the Quality Assurance Program description for the construction and operations phases. Next slide.

As I mentioned, we have one COL information item, and that's associated with Section 17.2, and that's typically Item 17.2-1, which requires a description of the quality assurance programs associated with both the construction and operations phases.

The UniStar QA program for construction and operations is documented in the UniStar topical report, UNTR 06-001 Alpha, Quality Assurance Programs Description.

The basis of that quality assurance program description, as required by Standard Review Plan 17.5 of NUREG 0800, is that the QA PD be based on today the 18-point criteria of 10 C.F.R. 50, Appendix

B, as well as basic and supplemental requirements and applicable subparts of NQA-1 1994. The QA PD was written to be compliant with both of, with these documents or these standards.

UniStar topical report, UNTR 06-001-Alpha, QA PD, Quality Assurance Program Description, Revision 0, was approved by the NRC on March 14th, 2007. Changes to the QA PD are and will continue to be made in accordance with 10 C.F.R. 50.54 Alpha (3), Conditions of Licensees, and 10 C.F.R. 50.55(f)(4), Conditions of Construction Permits, Early Site Permits, Combined Licensing and Manufacturing Licenses.

CHAIRMAN POWERS: I can't help but notice that you have unfortunate spacing here, because it says "QA PD as stated in the UniStar topical report, no."

(Laughter.)

MR. HARVEY: That was a --

CHAIRMAN POWERS: There is -- the no question I have. I mean this is said very glibly and appropriately. But when you look at that, with all of your experience in QA, those requirements in NQA-1, what are the ones that cause you the most challenge in implementing?

MR. HARVEY: Well, right now we have several challenges to us with regard to implementation with quality assurance programs. The first and probably the most obvious challenge is having individual staff that are knowledgeable and have the experience that are, so that they can in fact implement the quality assurance oversight functions.

We're working with EPRI, INPO, NEI in helping develop a -- working with development of a uniform standard for quality control inspection personnel, as well as quality assurance and NBE personnel with ASME. So really getting a qualified work force and being able to effectively implement the program would be my number one concern.

CHAIRMAN POWERS: I think that fits. Even more universally found is that it's just having the folks that understand not just the requirements, but the philosophy behind it.

MR. HARVEY: You're absolutely right. One of the big things that we're talking about with the development of that uniform standard is ensuring that the nuclear DNA, the 29, 30, 40, 50 year professionals in this room grew up with TMI and Chernobyl data especially, understanding really that -- the fundamental quality principles behind it, and the lessons learned behind those types of events, are incorporated into this new training program, so that when we do get people coming through these pipelines and addressing the new builds, I think we're going to be okay for the first couple of builds.

But as we get more and more plants developed, that's where we're really going to struggle finding those quality people, and ensuring that that nuclear DNA is incorporated into those training programs is crucial.

CHAIRMAN POWERS: Yeah, I agree with you, that manpower is going to be the limiting factor, and it's going to come very quickly. I know that Louisiana Services, just unqualified welders, having to train their own welders, because they just can't find them. That's going to get worse.

The other thing I know about these quality assurance programs, when they train people in those areas, they're very good about telling you what the requirements are. They're very poor at telling you, you know, why do we have this particular one? What are we trying to achieve with this program, with this particular requirement?

Since most of the requirements are experientially based, we don't have a good list of experience at why did somebody believe this was necessary. I mean it's because somebody failed some place along the line, some flaw was detected some place that had missed the thing.

You know, we don't have that good taste study, sort of training programs for people like yourself, that are trying to manage and develop a work force. It's not part of the regulatory process, but it's an interesting issue.

MR. HARVEY: No, it is, and we're working with -- there are four pilot schools, and we're helping develop that uniform curriculum that is really driving that, not just understanding what you're looking at but why you're looking at it, because that's absolutely critical.

CHAIRMAN POWERS: Yeah. We need a good taste study book on, for quality assurance. I mean there are just -- and it even gets worse when you move to, from construction to software sorts of things, where there are new sets of requirements and they really get obscure in that area.

MR. HARVEY: Yes.

CHAIRMAN POWERS: Please continue.

MR. HARVEY: Okay, thank you. Next slide please. All right. I just wanted to bring you up to date on some of the activities that we are performing from the quality assurance, quality control perspective at this point. We have qualified eight suppliers through full scope audits, including AREVA, Bechtel, EDF/Ceidre, as some examples.

We qualified EDF/Ceidre by full scope audit to perform our field oversight activities, specifically at Chalon and JSW, Japan Steel Works. We recognize, and I want to stop for a second. We recognize the importance of not just trusting someone, getting somebody else to be your eyes and ears.

So we review monthly reports from our supplier, EDF/Ceidre, to look at exactly what they're finding. But we're also spending the time and we're going to be going over to JSW and spending some time partnering with our Ceidre supplier and with JSW, to ensure that we have -- that we really have an understanding of our expectations, because we feel as though it's very important that we stay involved and we don't just rely on our suppliers.

CHAIRMAN POWERS: My personal experience in this matter, I was once at the Paul Scherrer Institute standing around, and to the people from Paul Scherrer Institute who had just taken GE's training in NQA-1, which is different from the European standards for quality assurance.

We were just coming out of a classroom, where GE was doing just what you already -- reviewing what their expectations were, and this nuclear quality assurance DNA you speak of, you know, they don't have it in Switzerland, because they come up on a different standard.

So they're very, very good at quality assurance. They come up under a different regime, and people were just angry. How can they tell us to do these sorts of things? This is the skill of the craft and things like that, and so you're right.

We have to go to great lengths with NQA, as we move to international suppliers, to communicate, and try to understand where they're coming from, because it's different. It's a very different regime that they come up with, even though I think we all recognize Swiss products tend to have tremendous quality assurance behind them. It's a different regime of quality assurance.

MR. HARVEY: I appreciate those comments. I couldn't agree with you more. I've spent quite a bit of time at some Japanese construction, plant construction locations, as well as spending quite a bit of time at JSW and some other European suppliers as well, and I think it's very important that we communicate and we have that constant communication and don't trust.

Because to be honest with you, in my opinion, ensuring that you have the common, the good communication, is you really have to have trust, and that takes communicating and that you can't just show up and beat your hand on a desk or something and walk out and expect something to happen. You've got to establish relationships, and that's really key in part of our job.

The last item I just wanted to mention is that quality and performance improvement personnel do have quite a bit of experience at this point, specifically with our RPV forging and fabrication, as well as several diesel generator fabrication and repairs and things along those lines. But we do have experience in the oversight of construction and fabrication activities. Next slide.

We do have one SER open item associated with the program, and that's specifically RAI 200, Question 17.05-6, which addresses each of the regulatory positions of Reg Guide 1.33. This open item is being responded to by Letter UN 10-106, and will result in a revision to the QA PD topical report.

We expect the revision to the QA PD topical report will follow very quickly, based on the NRC's review and approval of NEI 06-14, which is currently in the review process. We expect to issue our revision to the QA PD within 30 days following the final SER on the NEI 06-14 document. Next slide.

There's one SER confirmatory item associated with the program. Specifically, this action is to incorporate UniStar's response to RAI 120, Question 17.05-103, to remove redundant requirements of 10 C.F.R. 5055 Echo from the QA PD, and the SER confirmatory item will be addressed with UniStar Nuclear Energy revision to the QA PD to address this response in our next revision. Are there any questions?

(No response.)

MR. HARVEY: Okay. If there are no questions, the next section is Reliability Assurance, and I'll turn that over to Richard Szoch.

MR. SZOCH: Good morning. My name is Rich Szoch. I'm with UniStar Nuclear.

MEMBER STETKAR: Mark, you have to be careful with your paper on the mic.

MR. HARVEY: Oh, I'm sorry. I apologize.

MR. SZOCH: Hi. I'm Rick Szoch with

UniStar Nuclear. I've been with UniStar for two years. I've been with our parent company, Constellation, for 26 years. Most recently, prior to UniStar, I was assigned to the corporate office within Constellation. I oversaw the system engineering function for our five operating plants, and I spent 20 years actually at the operating unit at Calvert Cliffs.

Prior to that, I was with an architect engineer during the construction phase of the Comanche Peak and South Texas projects. I'm a graduate of Virginia Tech and a registered professional engineer in the state of Maryland.

I'd also like to introduce Gene Hughes at this point. Both Gene and I are going to present the overview of the RAP program. So I'm going to start with the beginning piece and hand that off to Gene midway.

I'll provide the overview of the program itself and how it interrelates with the engineering process, and Gene will present the details of the expert panel and the interrelationships with the PRA.

MR. HUGHES: To introduce myself, before we -- so we don't break this up as we go forward, my name is Gene Hughes. I'm a nuclear engineer from

North Carolina State. I've been working in the industry for 41 years and by an accident of starting late in the decade, I have practiced my craft in six decades. But I'm not that old.

I began my work at Duke Power Company, went to General Electric, where I spent close to ten years in the deterministic side, working in licensing. I made presentations to ACRS in '72 and throughout that decade, the most memorable being on a subject called "Potential Common Load Failure of SCRAM Systems," which was enjoyable and seems to have legs.

The subsequent years included some consulting. I developed the Limerick risk assessment that was submitted to the NRC in 1979, in support of the FSAR submittal. I subsequently formed Erin Engineering and Research, which I ran for 23 years.

After leaving there, I formed ETRANCO, which is a company providing services internationally, primarily associated with new build, but also supporting existing plants. I'm very happy to be the UniStar Acting Director of PRA.

MR. SZOCH: Okay, great. Thanks Gene. Slide 14 please. Okay. There are two COL information items I'd like to discuss briefly. The first has to do with the identification of a site-specific SSC list within the scope of the Reliability Assurance Program, the RAP. In our response to RAI No. 61, we identified that the U.S. EPR FSAR contains the design certification list, and that is, of course, incorporated by reference in RCOLA and the site-specific list is included in the actual Calvert Cliffs Unit 3 COLA, Table 17.4-1.

The second COL information item, I'd like to pause and spend some time on this. This has to do with the information request in Reg Guide 1.206, which basically describes the structure outline and function of the Reliability Assurance Program.

So to introduce this, we thought it would be very important to spend some time on how the program and the process actually works, from cradle to grave, from beginning to end, the beginning being now or in the design certification phase, and the end being through the operational phase, all the way up through decommissioning.

Primarily, the Reliability Assurance Program itself consists of two stages. Stage 1, the D-RAP, the design phase, and then Stage 2, the operational phase. So if we can go to page 16, Slide 16, these are the key elements of the Reliability Assurance Program, the blue blocks that you see there.

The expert panel on the far left, of course, is the key, the hub of the Reliability Assurance Program. The risk-significant list of SSCs, of course, is another key element.

We'll talk more about how that works and how that interrelates with programs included in the third block there, the QA/QC procurement, fabrication and construction installation testing processes, and there are others as well. That's an example of some key processes that are required to interface and rely on the Reliability Assurance Program. We'll talk more about that.

CHAIRMAN POWERS: I mean you correctly identified an expert panel as a key element of your program, and it's been one that we have raised questions on repeatedly. It's how do you select the experts, to have the depth that you need for the job? That is, they have to be fairly familiar with the plant you're actually going to build, and at the same time not have, for want of a better term, inbreeding, that is everybody operating from the same, out of the same building tend to have all the same kinds of views and things like that?

MR. SZOCH: On Slide 29, we actually go into detail on the expert panel make-up. So if you don't mind, could we hold onto your question, Dr. Powers, and then come back to that?

CHAIRMAN POWERS: We can definitely hold on. You may have to remind me of my question when we get there --

(Laughter.)

MR. SZOCH: Honestly, I think we'll hit it head on and have discussion.

MEMBER SHACK: I'll remind you.

CHAIRMAN POWERS: Oh, okay. Well I mean it is a question that comes up every time, and this -- I mean I saw Mark nod when I said "inbreeding." I'm not sure it's the right word, but he brought up DNA.

(Laughter.)

MR. SZOCH: Okay. So Slide 17, please. It's a continuation. As you can see, the maintenance rule and its interface with other key operational programs, is basically the core of the RAP program and the operational phase, as you can see. Let's go to Slide 18.

It's the operational phase, again being Stage 2 and the D-RAP, the design stage, being Stage 1. And on page 19 please. So as I mentioned earlier, the RAP program with those key elements in the blue box above, have interfaces with other key processes, design and PRA being two of them.

As you can see, the feedback loop that we provided in the blue box on the top there, with the Reliability Assurance Program, the block that lists the key processes, CAP, QA, procurement, etcetera, feed back into that risk-significant list.

What's inferred there is that each of those processes and/or programs are implemented, recognizing the importance of systems, structures and components that are designated on the RAP list. They're recognized.

So when they ask well what's different about an SSC that's on a list? They get special treatment. So for example, if it's a non-safety related component, there are still special treatments that are considered in each of these processes.

They're recognized in the same course as you go into the operational phase, and our operating plants do this today, through the work management process, the design process, through QA, warehousing, procurement. All of these components will get special attention commensurate with their safety significance.

CHAIRMAN POWERS: Then the question comes down, what metrics are you using for risk-significant?

MR. SZOCH: The risk-significant metric?

What do you mean by metrics themselves?

CHAIRMAN POWERS: Well, I mean you have something on what you call a risk-significant SSC list. There has to be some measure that you're using.

MR. SZOCH: Absolutely.

CHAIRMAN POWERS: To put them on a list.

MR. SZOCH: We're going to talk about that in a minute.

CHAIRMAN POWERS: Ahh. I'm such a great straight man.

(Simultaneous discussion.)

CHAIRMAN POWERS: --looking around on the payroll if I'm going to be a straight man like this for you.

MR. SZOCH: That's a key metrics used by the expert panel PRA, together with these metrics, and we'll discuss those shortly.

So the last point before we get into the makeup of the expert panel, we get into that, Dr. Powers, is I just want to talk briefly about the design process.

CHAIRMAN POWERS: Well, let me ask you a question before you get to that. Now you've told us things that get on this list or get special treatment, and you have an elaborate design of special treatment.

How do you know it does any good? How do you know that it does any good, or if they're adequate?

MR. SZOCH: Well, that's when the corrective action program comes in.

CHAIRMAN POWERS: Oh, you have an answer for everything, don't you?

MR. SZOCH: That is the -- that's the answer, and the corrective action program is very closely linked to a RAP component. So it would recognize anything on the RAP list, and that would get special treatment --

CHAIRMAN POWERS: And it tells you when you're inadequate. It does not tell you if you're doing more than is necessary.

MR. SZOCH: Well, through performance monitoring -- well, there's two answers to that. In the operational phase, you're going to have performance monitoring of everything in the RAP program. So you'll have precursors and you'll have identified precursors.

CHAIRMAN POWERS: Again, that tells you when it's failed. It's when you don't have --

MR. SZOCH: Precursors would be prior to failure.

CHAIRMAN POWERS: Yeah.

MR. SZOCH: Okay, so --

CHAIRMAN POWERS: But that will tell you that you're doing something that's inadequate.

MR. SZOCH: Right, or that you may be approaching failure. So you may have indications of component performance that may indicate --

CHAIRMAN POWERS: And those things I'll grant to you. What I'm asking you is what indication do you have that you're doing too much, you have in there what's needed, that you've got a --

Mark's given you some requirement that Mark felt was wonderful and it's just totally a lot of labor, a lot of work and it's just a waste of time.

MR. SZOCH: Okay. Well, we have these -- again, we have these processes in place today in our operating units, when we execute the maintenance role, which is very similar and we'll model that process. So if you find you're doing too much, and that's where maintenance comes in.

Maintenance will go out and provide an observation of the condition, and provide that feedback, either through the work management program, the maintenance program and/or the CAP program, which will be tied together.

If we're looking at touching something too frequently, and that sometimes can be more of a degradation than a help, over-calibrating, over-maintenance. As you know, many events in the nuclear industry are caused by maintenance events, so you don't want to touch it too often.

CHAIRMAN POWERS: Again, personal experience, where we were testing diesel generators at Rock Flats. The manufacturer suggested doing it every six months. The DOE area office said well, if it's six months, do it every three months, and of course the contractor said well, to make sure we make that, we'll do it once a month. They were burning out the diesel generators from testing them.

MR. HARVEY: Right, right. Well, diesel generators weren't made to be operated in that way. They're made to be started once and --

CHAIRMAN POWERS: That's right, and that's exactly what was killing them.

MR. HARVEY: Actually right. But just, and I apologize, but there are standard PM optimization programs out there that are geared towards doing the right amount of maintenance. So that's what we were looking at, is utilization of a PM optimization program.

CHAIRMAN POWERS: That's what I was

looking for. Good.

MR. SZOCH: And that of course is going to be standardized program that, you know, interfaces with EPRI templates. In fact, we're starting to implement that PM program now as we speak today. It's part of the design process early on.

CHAIRMAN POWERS: That's exactly what I was looking for.

MR. SZOCH: Okay. So speaking of design, the importance of the design process in the RAP requires feedback between the two. So the design process, of course, will evolve with time.

As we get more detailed information, as design either evolves or changes, it will feed back into the Reliability Assurance Program. So there's constant checks and balances between both the design and reliability assurance process. Okay, Gene.

MR. HUGHES: Before we change the slide, let me cover a few things. Let me start with your first question. Your first question, to remind you, and I know you remember it, was the make-up of the expert panel and how do we know they're qualified.

This panel and this process that we go through, is one that operates by procedure. It operates by practice. So we have to have qualifications identified, and we identify the people and make sure they have training, that they understand the process that they are following, and they're able to implement it.

So there is a standard process to go through. The key thing I would identify is the expert from maintenance is not an expert because he is a person who spent 20, 30, 10 years in maintenance.

He's an expert because he's a person who interacts with the maintenance practice for the plant, the maintenance expectations at this stage, but later in operations the actual occurrence.

He comes to the panel to bring all of the information that Maintenance can bring to bear on the correct decision. Which brings me to your second question, which was the metric.

The metric for a risk-significant SSC is not so much a critical determination in one area or another, as it is the integrated consideration of those people and those organizations that bring together an understanding of the risk significance as interpreted by the PRA, the risk significance as interpreted by the PRA person that's beyond the scope of what's in the PRA, the maintenance information, the operational information.

All of that together goes into the determination of the risk significance of the component. I'll describe that in a little more detail in a second.

Let me point out one thing about this particular slide, that we were only able to get so many dimensions on this. The RAP program across the top shows the three boxes. Those in fact do not occur once in time sequence as shown, but occur over and over and over.

They transition at fuel load over to the operational phase, and the maintenance rule is an example of that. The design process shown here, on the other hand, is really a left to right in time. So what we're looking at is today, we have the initial design and the expert panel has met two, three, I think maybe four times.

The expert panel that has met has been put together from people at AREVA, a partner and a member of the consortium for the project, and at the request of UniStar, AREVA has conducted these meetings, making sure they had people that understood these disciplines and could bring to bear that information.

MEMBER STETKAR: Gene, you're going to get eventually to that slide, whatever the heck it is. Is it better if we hold off specific questions until we get to each topic, so that we can get through this general flow? Because you're starting to get into some of the details that I wanted to ask about, but I don't want to interrupt the flow of the process.

MR. HUGHES: Well, I'm happy to proceed and get there quickly.

MEMBER STETKAR: Yes. I'm just asking you which way you think it's better to address these things.

MR. HUGHES: Probably let me get a couple of slides further, but I've been to ACRS too many times to try to hold you off too long.

MEMBER STETKAR: But on the other hand, if the ball is rolling, you know, if you see a stop to it, that would be good.

MR. HUGHES: Oh. What I want to make sure is clear at this particular slide is the work that's been done to date is primarily at the system level. As additional information becomes available, it will be refined to the component level, and for the component to the functional failure that is particularly significant, and we can go to the next slide.

The expert panel, I mentioned the

function, the criteria that are described in the FSAR. The criteria are fairly well described and describable from a PRA perspective, but in fact they have to include qualitative understandings from maintenance operations, the deterministic side, and those are somewhat less numerical. But they come together to reach a consolidated opinion as to the significance of the component.

The PRA insights, maintenance considerations, those are the activities that are brought to bear, and I can go to the next slide.

What comes out of this significance list is more than a list. It is a list that includes what is or isn't in, what the functional significance is, the thinking that went into that, and so it's a real documentation of the information that's needed by procurement, fabrication and construction, to pass that information forward.

The function here is to create the correct decision and judiciously carry it forward in a documented manner so it can be used. Am I getting close to where you wanted to --

MEMBER STETKAR: No, you'll eventually get there.

MR. HUGHES: Okay. One more. Go to the

next one.

MEMBER STETKAR: I think it's probably from, I don't know, Dana. From my perspective, if you have several slides through number 28 that show sort of the flow and the interrelationships, it's probably better if you --

MR. HUGHES: Just get through it all.

MEMBER STETKAR: Get through that, and then we'll pick up details and specifics.

CHAIRMAN POWERS: Well, let's go back one slide.

MEMBER STETKAR: We're not.

(Laughter.)

CHAIRMAN POWERS: I'm going to go ahead and interrupt the flow, because on Slide 21. If we can go back to 21, I have a question on that.

MR. HUGHES: Oh yes.

CHAIRMAN POWERS: Okay. We have PRA insights, we have maintenance considerations. We have something called defense in-depth or we continue to struggle with exactly what we mean by that. We have safety margins.

MR. HUGHES: Let me go through all of them.

MEMBER STETKAR: Yes.

CHAIRMAN POWERS: Well, as you go through them, suppose that I have a PRA insight that says that I do not need a redundancy or a diversity, that defense in-depth say I do need. How am I performing the integral over all these things?

MR. HUGHES: Okay. I think I will trip through that, but I'm sure you know how to stop me. Let me start with defense in-depth. Defense in-depth is a process we use, and we have deterministic criteria documented in 10 C.F.R. 50 and 52, and those criteria set forth a postulation of transience accidents, and we do things to demonstrate compliance.

We come out of that analysis and that process with what I would call defense in-depth, in that it includes redundancy, it includes barriers, it includes challenges to those barriers. Coming out of that, we have a safety grade determination of what the component safety grade performance expectations are.

That information comes in and the people that bring that information have to have knowledge of those calculations and how they're done and what's credited in the analysis and what's credited in the safety analysis report.

We then look at the PRA. The PRA has in it an analysis that assumes success or failure of components, and the PRA exhibits our knowledge, the knowledge of the PRA team as to how the plant and components and systems function.

In this discussion of the panel, it is entirely likely that the PRA team member will from time to time gain insights into a greater depth of understanding from either the maintenance individual or from someone else in the group, as to how the components function together, which may reflect on the information he brought in the first place.

If I can step aside for a second. In the significance determination process, what we often find in operating plants today is the PRA treatment is insufficient to capture the particular scenario that we're looking at. So we can refine that, and this understanding is similar to what happens here in the panel.

The safety margin is the effort to make sure that we aren't creeping up toward a limit. A good example of that is a transient calculation in which we have limits related to departure to nucleate boiling, as defined one way or another, and we don't want to be creeping ever closer to what those limits are.

We have the potential for something

causing a transient. That's a safety input to this process, and in evaluating the component, if we determine that the likelihood of a transient could go up unless we do certain things with it, then that would be factored into the determination that that information should be carried forward, and it should be put into the program to assure that it's correctly treated.

The technical specifications is a subset of information that is critical, as identified through the deterministic licensing activities, which incorporates the defense in-depth as its underlying fabric.

So that's the structure that we bring together here, is this group of people to evaluate, bring to bear these considerations, make a determination, but perhaps more importantly, document the basis for that determination. I'm not sure I answered your question, so I'll stop a minute.

MR. HARVEY: Dr. Powers, I'm sorry. I just wanted to build on something that Gene mentioned. To go off of that, some of these items are looked as yes/no, did I meet a certain criteria, yes. It doesn't matter what other ones meant. If it meets any one of these tech specs, safety margins, whatever, then it moves over and it is moved over into the safety significant rank.

You have others where, exactly as Gene said, you may have your PRA, PRA insight individual, where based on information that you're getting from maintenance considerations or how close you are on your safety margin, that may move up in aggregate.

So I think you're looking at why we have a panel. It goes to why we really have a panel, because it can't be a black and white. You've got to use a collective intelligence that all of this is bringing together.

MR. HUGHES: There is -- this is good dialogue we're having, if you don't mind. There's a great example of this, the diesel generator. The nuclear plant has in it 50 plus thousand components. PRA has 3,000. One of the ones in the PRA is the diesel generator.

The diesel generator is an entire power plant. Of those thousands of components in the diesel generators, certainly some of them are critical to start, load and run. Some of them are critical to periodic check a meter, to see what an indication is. We don't need that in order to start, load and run.

So the safety significance of the

components within that engine can easily be differentiated into different groups. The PRA brings the knowledge that it's start, load and run that's critical. The deterministic defense in-depth brings the knowledge that it's load that's critical. Load, start, load and run. Then together, the group can identify those components that are important to achieve those, and the failure modes that would challenge the success of that.

CHAIRMAN POWERS: A very useful example. That's a very useful example. I'm getting a sense of how you do the integral. I'm sure that there are other examples that will come up.

MR. HUGHES: One question that I'm sure you'll ask at some point is what if this panel doesn't agree? That can happen, and if the panel doesn't agree, then two things can occur. The representatives can go away and think about it and come back, after they've taken in what they observe, or you can go to plant management, senior management and have it out.

So there is a process if you are not in agreement to achieve resolution, and then the thing that this shows with the arrow coming back from procurement, fabrication, construction, and there would be an equivalent arrow coming back in operations.

If we find out that there's something different, this panel continues to exists, and through operating feedback, through the MSPI program, once the plant is operational, through the three year PRA update for plant-specific information, through the maintenance rule, which is triggered by failures that may occur, all of those can lead to reconsideration and to steps to enhance and correct whatever might have been missed.

CHAIRMAN POWERS: A thought just came to mind. This process, from where you start through fuel load, it's wrong, and episodically you're going to replace members on an expert panel. Having looked forward, each member on the expert panel has a particular area of expertise. They're not a bunch of generalists. They're there for a purpose.

You get a guy in. What if he disagrees with his predecessor? Well, a maintenance guy says oh, that's not the way we -- that's just not the way we do maintenance here.

MR. HUGHES: I think the answer would be that would be a problem.

(Laughter.)

MR. HUGHES: The solution to that is to

address it ahead of time.

MR. HARVEY: Well, it's a resolution process. That's why we build processes to control well.

MR. HUGHES: But bearing in mind that the representative from Maintenance is not an individual. It's not Joe. It's the representative from Maintenance, and Joe is supposed to be going back, meeting with his peers, discussing the decisions that are being made.

They are seeing what's coming out of this process and if someone that would be in a position to replace him, who would be in the more senior level of that organization disagrees, they should be hashing that out well before that person steps into the panel.

CHAIRMAN POWERS: Okay. You were very careful to say he's supposed to be. Does he?

MR. HUGHES: That's why we have those programs. Hopefully, yes. This is not --

MR. HARVEY: This is a living process. I mean we all know that if we have -- if it were just once, we meet once and we have this list that's going to live forever, and we're not going to revisit.

We're not going to gain new information. We're not going to change our maintenance practices which may impact this. Then we wouldn't even need the expert panel to continually meet.

That's why we have an expert panel that's going to continually meet, because we expect those changes as the design changes, as we get further in construction and as we operate the plant.

MR. HUGHES: The other factor that would probably somewhat address the potential for this is the fact that it's a panel. It's not a person. Then you'd have your issue in spades. If it's a panel of two, that might be a problem. If it's a panel of several people who have thought through this process together and reached a decision for the organization.

CHAIRMAN POWERS: I presume that as you go from left to right, that the panel probably gets bigger.

MR. HUGHES: No. The panel hopefully begins with all of the organizations represented. Right now, that's a little problematic. We don't have an operational organization. So UniStar has turned to AREVA, that has expertise in this area, to have someone on the panel that can represent that thought process.

We don't have a Maintenance organization that's performing the maintenance. So UniStar has asked AREVA to provide someone with that background and that experience, so that we can try to capture that thinking up front. But certainly as the plant goes into operation, we would have people with those expertise areas.

There would be some refinement as we go into operation, because the panel may shift a little bit. There will be refinement as UniStar steps in and takes more direct control of this, rather than relying on our consortium partner, AREVA. But those will be refinements. They won't dramatically change the make-up of the panel.

MEMBER RAY: Gene, the output from the expert panel is input to this thing called the risk significance structure system and components list. That's their sole output?

MR. HUGHES: Well, it's the combination of the list and the little box I've got above, that identifies the disposition, the significance, the reasons, how they took in the PRA input, the deterministic input --

MEMBER RAY: Okay. Well, bear with me a second, because you've got a feedback here from something called procurement fabrication construction. To me, that feedback loop is just full of dollar signs that says no, this is too expensive. It's impractical. We're going to do it a different way, because ultimately there will be a budget that has to be met.

I guess the question that I have is as long as some objective limit is met, if you've got a way to reduce the cost, you're going to do it. At least most people involved in turning out these things I've ever met, that's what you do. Am I right about that?

MR. HUGHES: May I use my words? My words would be that the individuals involved in installation or in procurement may encounter difficulty, and the difficulty they encounter could be high cost. It could be an unavailability --

MEMBER RAY: Let me guarantee you something. It's going to be high cost.

MR. HUGHES: It could well be.

MEMBER RAY: Right.

MR. HUGHES: If it is, that information --

MEMBER RAY: And it is, and said -- these are my words. "When it is."

MR. HUGHES: I respectfully appreciate your words, but should that information be fed back, the panel's job is to answer the question, is there an alternative to this or is this what we need to do? That's what the panel's function is.

MEMBER RAY: I don't see that arrow on here, which is why I asked the question to start with.

MR. HUGHES: Yes. Well, I took the arrow back to the risk-significant list, only because subsequent arrows --

MEMBER RAY: But one thing tells me that when one of these projects actually gets moving, this is going to be a very -- a process in which the panel's overwhelmed by all of the changes that need to take place to reduce cost.

And I guess you've answered the question, that they'll get involved somehow. But that's -- everything becomes theory until you actually go out and buy and build and hire and fix things that don't work out the way you thought they were going to.

So I'm more interested in how this process works in that domain than I am anything else. What is the objective limit that says "No, by God; you've got to go ahead and pay what you don't want to pay to do this"?

MR. HUGHES: Well, the limits -- I'm not sure I have a good answer, but I'll attempt. The various individuals that you bring into the expert panel represent the commitments, decisions, activities that have been made. If the PRA information suggests that failure to do something would result in the core damage frequency increasing, then that would be a significant issue to be addressed.

MEMBER RAY: Well, it's going to increase. So what? I'm asking you how this actually works in the real world, because the real world is going to be I've got a cheaper gizmo or I've got a cheaper way to do something, and we're way over budget. I've got to do it. Now how does the process work?

MR. HUGHES: Well, if the cheaper widget that you describe is one that violates any of the commitments in the FSAR, or violates --

MEMBER RAY: What do those commitments entail in this domain here of reliability assurance?

MR. HUGHES: Well, all of the risk-significant components that come in as a result of being safety grade are --

MEMBER RAY: So this is in the Part 50 space. I'm talking about things that are -- that result in this panel that Dr. Powers has been asking you about. They've got, they've driven the risk way down further than it needs to go, and I'm over budget. What do they do now?

MR. HUGHES: I don't think they sit around and figure out how to raise the risk.

MEMBER RAY: No, they don't. But they're confronted with a lot of changes that have to take place, in order to reduce the overrun that I'm looking at. If this is too hard a question, we'll skip it.

I'm just trying to figure out how, because in the real world, you've got to implement this stuff, and unlike the government programs, you're not going to wind up with just do more of it until everybody's happy.

MR. SZOCH: We actually face this problem today in the operating units.

MEMBER RAY: Of course you do.

MR. SZOCH: And we have this question all the time.

MEMBER RAY: If you're licensed under Part 50, which is a little different maybe.

MR. SZOCH: Correct, but we still are trying to reduce cost and that question, your question is real today. It happens.

MEMBER RAY: I know.

MR. SZOCH: So the answer is, and the way we run it today and the way I see this running is in a very similar way. Economics and cost is really not part of this decision-making process. It's an input. But this panel needs to stay pure to the objective requirements of the program, and it will. However --

MEMBER RAY: Now that makes sense, and I think you're right.

MR. SZOCH: How do we address your question.

MEMBER RAY: It's pure theory.

MR. SZOCH: It really has to work that way. Now there are separate panels out there that look at budget, and they'll challenge these guys --

MEMBER RAY: They've got a bigger stick.

MR. SZOCH: And they've got a bigger --

MEMBER RAY: Yes, they do, unless you run into some absolute limit, which is what I was asking Gene about.

MR. SZOCH: But the answer isn't always to cut this list back.

MEMBER RAY: I know that.

MR. SZOCH: The answer is --

MEMBER RAY: I'm not trying to set up a straw man here. I'm just saying at the end of the day, if you keep this panel pure, as you said, then it's advice. But by God, if I've got a budget to meet and I can do it in some other way that doesn't violate my commitments, which I can't violate of course, and I'm trying to figure out how they get engaged in that, and you're trying to answer that, so good.

MR. SZOCH: Well, how about an example?

MEMBER RAY: Go ahead.

MR. SZOCH: We tried to standardize this very process in our operating fleet of five reactors. So we got really three different sites with three totally different processes before we purchased Ginna and Nine Mile Point and tried to align those with a standard process.

Cost overruns were at different degrees with the different plants. The idea is to get them all number one standardized, the standardized approach and then streamlined. What we have found is we weren't doing the right work and paying the right attention to the right stuff. We were maintaining things that we really didn't maintain. We were putting design controls, quality assurance, even corrective action program on everything.

Treating everything important and treating everything with significance at a high level isn't economic. Where we found where we can cut back is keep this as pure, but look at everything else that's not on the list.

Do you really need that same level of controls in all of the different processes that we have today, and we found that we didn't. That's where the cost savings are. We need to ensure that we apply the same --

MEMBER RAY: Well, let me just -- I'm taking up too much of the committee's time now maybe. I think that works in Part 50. It may work in Part 52, but we ought to try to understand how this is supposed to work in this new world that we're talking about here. I won't take up any more time. John's got a question. But you helped, okay.

If this is a process that is divorced from, I'll call them management decisions required to actually build and operate something, that's fine. I understand how that works. This is advice, take it or leave it, but it isn't something that's enforceable, except that's where I was trying to get Gene to tell me how you enforce this. Want to try one more time, Gene?

MR. HUGHES: I'm having a hard time with the question. The expert panel of the reliability assurance program is a program that's put in place that's a regulatory commitment, and it's measured and monitored, and its role is to perform this function.

MEMBER RAY: That's right, and so if you decide to make a change that increases a risk, they'll reflect that. As long as it isn't excessive or unreasonable, you'll go ahead, right?

MR. HUGHES: I would hope, and I would expect that the panel would seek to find ways to avoid increasing the risk, but to achieve the objective.

If it in fact came to a point at which, as you postulate, the choice was either increase the risk by a significant amount, a factor of two say, increase the risk or suffer a very large expenditure, I believe in every case I've seen they would be able to find ways in which they could achieve the objective of addressing cost without increasing the risk substantially. I think that's the history we have. That's usually what we're able to do.

MEMBER RAY: Well, I hope you stay around long enough to see whether it's borne out in real life. Okay.

MR. HUGHES: I intend to try.

CHAIRMAN POWERS: The difference is all your experiences with plants, they have risks, CDF is around ten to the minus fifth, something like that. Now you're working with a plant where the risk metric, I don't exactly know what it is, but it's roughly around ten to the minus eighth, seventh, something like that.

And now all of the sudden, some things -- it's not a factor of two. Suppose I increase the risk to ten to the minus sixth? I mean I can go to Reg Guide to 1.174 and find out that's not right, that's fine.

MEMBER SHACK: Now John's going to have a new subcommittee meeting on Risk Metrics for New Reactors.

CHAIRMAN POWERS: That's fine.

MR. HUGHES: I would, at the risk of something, I would be happy to address your comment and the concept of risk metrics. Risk metrics are somewhat problematic, because we do a calculation and we generate a result which we have confidence in. We learn new things. We adopt new practices. We adopt new approaches, and the number may change some. But the plant is still the same plant.

As we go forward, I think the key thing coming out of the risk assessments is the understanding of the risk insights, the importance, the significance of components and systems, rather than that absolute value of the number.

What comes into this program is that

understanding of the significance of those things. I wouldn't expect that absolute number to jump all over the place, but it is going to change over time.

There are going to be different approaches. We're looking right now at approaches to doing fire PRA that are generating different perceptions of what the number is, and there are arguments about how conservative that may be.

I'm not going to try to get into that. But my point would be certainly these numbers will shift over time. But that doesn't mean the plant isn't steadily getting better.

MEMBER SHACK: No, but you were talking, I think the context of making a change that would actually -- a physical change in the plant, and in Harold's case, you're changing a component by some way.

MEMBER RAY: Yeah.

MEMBER SHACK: So we're not talking about the kind of change you get the PRA because somebody used a new method. We're talking about a change to the plant which reflects a real change in risk. But I think this whole thing, you know, I listened to this discussion and it seems to me, you know, we still have this --

The one thing that is pretty clear, you guys have a process for picking out the risk significant components. It's not at all clear to me that the same expert panel are the people who, I'm looking at NUREG 0800, and this is going to be a question for the staff, where one of the acceptability criteria was you "verify the acceptability of procurement, fabrication and test specifications for SSCs, so they reflect the reliability values assumed in the PRA and the quantities in the deterministic analysis." I was going to say okay, who makes that decision?

MEMBER RAY: That's right, Bill. They got this cheaper thing.

MEMBER SHACK: To me, it looks like I have two panels. One panel says this is important; the other panel says "Okay, this is what I've got to do to meet that reliability requirement."

It doesn't seem to me they're the same panel, you know. They're really different questions, and how you answer one question is different than how you answer the other question.

MR. HUGHES: That question becomes very significant if in fact one of the components in the plant has been evaluated in the risk assessment and in other ways as having special capability beyond what is normally available.

We do rely pretty heavily in the PRA on generic information, on reliability values that have been achieved, and a component that would have a significantly different reliability would be something that hopefully we would identify and be able to avoid.

MEMBER SHACK: Okay. So the answer is you order the thing like you ordered it the last time for something that was important.

MR. HUGHES: Unless we have identified that it might benefit, and it would be appropriate to include additional quality assurance inspection or other activities. Those would be identified and that could be undertaken.

MEMBER SHACK: And the other thing that sort of struck me as puzzling is you went through your whole discussion of the expert panel without talking about the process that's sort of described in here in some detail, where you ask your five questions.

(Simultaneous discussion.)

MEMBER STETKAR: I'm hoping that we'll get --

MR. HUGHES: That's next.

MEMBER SHACK: But that didn't sound at

all like the discussion you were talking about from the expert panel, you know. I'm assuming the expert panel really does go through this.

MR. HUGHES: Yes.

MEMBER SHACK: Okay.

MEMBER STETKAR: There's two slides that if we ever get to them --

CHAIRMAN POWERS: If we ever get there.

MEMBER STETKAR: --that we'll probably spend an hour on those two slides.

MEMBER SHACK: That's okay.

MR. HUGHES: May we go to Slide 22, jumping ahead?

(Laughter.)

MR. HUGHES: 22, I think we've covered. It points out that the design evolves over time, and that information is fed back.

23. 23 brings in the various stages of the PRA, and at the design certification, the PRA that's being performed bounds the plant-specific PRA.

Subsequent efforts will be undertaken to enhance that combined operating license base PRA with greater plant details, leading to ultimately as the as-built, as to be operated PRA prior to fuel load, that meets the standards in effect one year prior to fuel load, and then transitions to the PRA to be maintained for the life of the unit and updated from time to time.

Can we go to the next one? What this slide seeks to pint out is simply that there is interaction between the PRA and the design. It doesn't occur at discrete steps. It occurs continuously. Next slide.

And the feedback from the design to the expert panel comes, in the next slide, from each of these various activities, the next one and the next one. So that the panel is recurring in its efforts to meet, and I think at this point we might be ready for John's question.

CHAIRMAN POWERS: I think what we'll do, with his promise that we're going to spend two hours on this slide, we will take a break for lunch.

MEMBER STETKAR: Well, it's two slides, an hour per slide.

CHAIRMAN POWERS: An hour per slide. I stand corrected. I stand corrected and look forward to it. So we'll take a break for lunch until five minutes of one, and thank you very much, by the way.

In some cases, we're kind of going aside, but this is a useful discussion for the subcommittee, and we very much appreciate you laying out the issues for us to discuss carefully. So break until five of one.

(Whereupon, at 11:51 a.m., a luncheon recess was taken.)

A F T E R N O O N   S E S S I O N

12:51 p.m.

CHAIRMAN POWERS: Okay. We are reviewing the Reliability Assurance Program, and we have gotten to the infamous Slide 29, and Gene, I think you get the introductory comments on this. After that, I have no control.

MR. HUGHES: Fair enough. I'm pleased that we had the lunch break, because it gave us a chance to discuss this among ourselves.

CHAIRMAN POWERS: Which you had not previously done?

MR. HUGHES: Which we had previously done, but it's just to remind yourself of what you had covered. Let me cover a couple of things from this slide. I think I've covered the membership pretty well.

CHAIRMAN POWERS: Will you promise to explain to me how you get breadth and depth, without getting inbreeding.

MR. HUGHES: Boy, we didn't get far.

(Laughter.)

MEMBER STETKAR: How many members were on this expert panel?

MR. HUGHES: So far to date, the three or four that we've had included, I believe, and AREVA conducted them for UniStar. I think it was five or six members, am I right?

MR. TALLEY: No, nine.

MR. HUGHES: Nine.

MR. TALLEY: Charles Tally, AREVA Manager of Engineering Integration.

MEMBER STETKAR: Are all of these AREVA employees?

MR. TALLEY: Yes sir.

MEMBER STETKAR: Okay. That answers the inbreeding question.

CHAIRMAN POWERS: Inbred.

MEMBER STETKAR: Inbred.

MR. HUGHES: Now may I comment?

MEMBER STETKAR: Well, let me ask why, if this is a panel that is developing a site-specific list for Calvert Cliffs, why there are no Calvert Cliffs members on this panel, and why there's nobody from outside of either the design and operating organization that might bring to bear other insights or experience that perhaps AREVA and, you know, let me characterize this, the AREVA trainees don't bring to the table?

MR. HUGHES: Yes. The process itself will include in the future direct participation by UniStar personnel, independent from the AREVA personnel, and will include broader representation, although in a narrower, smaller group would be my expectation.

But I did want to comment, so please let me do that. I've been asked on several occasions should Bechtel be represented on the panel, and I would draw your attention to the bullet that says "Designated individuals," excuse me, "having expertise in the areas of."

It's not a panel of organizations, and the information brought is not AREVA or Bechtel or UniStar. It's the understanding of the technical work, and that's what the people have to be trained for. But your point is well-taken, of having diversity being a very appropriate thing.

MEMBER STETKAR: I was interested more not diversity of the organizations, although I do have a bit of a concern for everyone being only AREVA, and having only the AREVA design experience, and if they have some operations and maintenance experience, it's still within the context of their current operating fleet. I don't want to mention whether it's architect or engineer.

I was more concerned about having a

broader perspective on operations, maintenance from other organizations that have operated and maintained nuclear power plants. You said well, you didn't have any -- you obviously don't have any direct operations experience at UniStar with this particular design.

But on the other hand, even within your organization, you have a heck of a lot of operating experience, and a heck of a lot of maintenance experience to -- their pumps and pipes and dials aren't particularly different than anybody else's pumps and pipes and dials, and developing a program to keep them maintained and ensure their reliability shouldn't necessarily be all that much different.

Or identifying the types of systems, for example. When I think of populating the RAP list, developing the types of systems from other people's experience that might be important from their perspective.

MR. HUGHES: The FSAR includes a statement that the panel in the future, at a date not specified, will include direction from the UniStar Vice President of Engineering, and there will be UniStar personnel included in future meetings. But that is a function of how the project proceeds and when those organizations can be pulled together.

MEMBER STETKAR: Now a question, though. In terms of the COL licensing process, when in the future will that panel be convened, because as part of the COL, you do have a populated RAP list, and it would seem to be prudent to have input from that panel for the final RAP list, that is indeed submitted as part of the licensing application.

MR. GIBSON: Let me go ahead and address that. It will be implemented soon. It's being implemented in conjunction with the evolution of the design. So now we're getting into more detailed design, and it's getting down to the component level.

We have an ops and maintenance organization within UniStar now. So without pinning it down to the date, I would estimate within a year, plus or minus six months, roughly based on a current milestone schedule for our design development. But that process would be place including procedures, the panel, and will be expanded to include that UniStar membership.

MEMBER STETKAR: I guess what I'm asking, and I've forgotten the time line on the COL license application approval process, will that panel be convened and have an opportunity to feed into the RAP list in a timely manner, before the list is actually accepted as final?

MR. GIBSON: On a time schedule that Surinder put up, which shows us completing the Phase 6 in July of 2012. We assume a hearing process thereafter.

We feel very confident that between now in July of 2012 that we will not only have the expert panel put together, but we're going out for the development by that time of a full scope internal-external Level 2 PRA, including fire and flood, and we're in the process of developing that. That takes over a year.

MEMBER STETKAR: It takes some time.

MR. GIBSON: It takes some time. But we are in the process of moving forward, and --

MEMBER STETKAR: I guess, you know, I'm just worried a bit about the relative timing of these things, beginning into the staff schedule for getting approval of things.

MR. GIBSON: I'll ask them about more of the scheduling.

MR. HARVEY: I just want to point out. We've been talking about this internally at UniStar for an extended period of time, with our partners at AREVA. While we've used the term there AREVA's performing this function for us, they're not really performing this function for us.

They're required to have an expert panel based on the design and work that they're doing. We're taking advantage of the fact that AREVA has the expert panel, and that they're doing, they're appropriate based on their design, their doing their expert panel work.

Now what we're doing obviously is we have these plans from the future, and what we've got to do is develop a process to ensure that we build off of, okay, this is where we are in this stage.

Now what do we need to do to take that site-specific work and build that, because obviously the AREVA partners in this can't be doing a lot of the site-specific activities when we get down to the component level in more detail.

MEMBER STETKAR: That's right. I'm not, you know, and I think it's good. You obviously have to build off the expertise that you have available. It's just that we do have experience in the industry, where designers know an awful lot about the design and the design criteria.

They aren't necessarily as well-versed in -- they certainly don't have the appreciation for any plant-specific changes that you might be implementing in the design, nor in some cases do they have the integrated operations-maintenance type of perspective, even though they may have people on their panel that have those nominal qualifications.

That was more my concern. I wasn't trying to imply that it's not good to build off their expertise.

MR. HARVEY: We have the same concern.

MEMBER STETKAR: Okay.

CHAIRMAN POWERS: One of the favorite questions that one of the newer members of the Commission have, often has on expert panels is how do you decide on the qualifications? If a man comes before you and says I've been working in this field for 29 years, it's entirely possible he's been wrong for 28 of 29 years. So how are qualifications addressed for this panel?

MR. HUGHES: Well, it's a combination of experience, coupled with the direct awareness of the individual and the decisions they've made. The people on the panels thus far were selected by AREVA based upon their experience with those individuals and their knowledge that they would represent judgment in a proper and fair manner.

Going forward, the individual from Maintenance, as an example, will be someone within the organization who has garnered a level of respect within the organization. It wouldn't be a new individual or an unknown person, or the least among many.

It would be the person that had the confidence of the organization and that organization's management that they could represent Maintenance well. That's a judgment, but that is --

CHAIRMAN POWERS: Did you vet them? Did you vet the choices? Did you vet the choices? In other words, does AREVA says "Okay, well we picked this guy," and you said "no, no, no. No good for me."

MR. SZOCH: No. To this point, we have not. It's an AREVA-controlled process. We have oversight for the procedures and we understand the process and procedures that are familiar and we've accepted those. I mean we haven't challenged the individual qualifications.

CHAIRMAN POWERS: But you could if you wanted to?

MR. SZOCH: We could, absolutely.

CHAIRMAN POWERS: Is there anybody on the panel that you have --

MR. SZOCH: Yeah, we are aware of the panel membership. We work with them, both individually and in that capacity.

MR. HARVEY: And we've done independent assessments of their program.

MR. GIBSON: And you have no concerns?

MR. HARVEY: No concerns.

MR. SZOCH: And we're anxious to add our own membership to the panel, with our Operations and Maintenance folks. They're chomping, they're literally chomping at the bit to become part of this. They have experience with similar panels from a maintenance rule perspective on operating plants.

So we have a vision of exactly not only of the types of backgrounds, but actually the individuals we intend to insert on this panel in the near future.

CHAIRMAN POWERS: Yes. I have to admit I worry a little bit about corporate culture, because I can't believe that your corporate culture is identical AREVA's.

MR. SZOCH: Right.

CHAIRMAN POWERS: So I suspect there will be a transition period. The gears may grind a little bit.

MR. SZOCH: Absolutely.

MR. GIBSON: Let me also add that one of the strengths that UniStar has is of course our parentage with not only Constellation but Electricite de France. Electricite de France has the largest database of equipment reliability in the world. They also were one of the designers.

In fact, Christian Clement, in the audience back there, is one of the fathers of the design of the EPR, who works for us at UniStar, and we have our expats who are part of this.

We can draw on a very wide range and a large scope, a larger pool than would be typical for the traditional operating fleet that we're kind of used to. So we have a lot of assets that are very unique to the EPR and to our relationship, and so we're really looking forward to having not only UniStar but also the UniStar EDF components.

MEMBER STETKAR: Greg, I hate to say this, but I think just for the record it's important. You mention things that EDF has, you know, the largest equipment reliability database in the world. But in the sense of Dana's comment, EDF really hasn't shared that data with anyone. It hasn't been vetted by the international community.

The French, I hate to say this, but the

French tend to keep their data and their analyses to themselves, and therefore although there might be a large volume of material there, it's not -- it hasn't received an independent verification from outside of that organization.

I think that's a little bit of what Dana was mentioning, in terms of independent --

MR. GIBSON: I understand that they've had several examinations, worked with Gary Holohan and other people within the Commission with regard to ASN. You're right. ASN has been the primary focus because up until now, EDF has been primarily French But now they're becoming a global.

MEMBER STETKAR: Yes.

MR. GIBSON: And so within that global community, the EPR family worldwide is very extensive, and I think you will see that changing over the future.

MEMBER STETKAR: You know, I think a little bit of what both Dan and I have been saying is that it would benefit confidence in this process to have a bit more, you know, input from, directly from your owner-operator organization, and even some external independent --

(Simultaneous discussion.)

MR. GIBSON: Just to give you --

MEMBER STETKAR: The information might be perfect, but there's always that concern that --

MR. GIBSON: As you know, that's what we're starting on right now. We only recently had the acquisition. But the future looks very bright for that, and I wouldn't want to speak for one of my parents, EDF. But it looks very bright and our vision is truly one of a global community.

MR. HARVEY: And we do have two individuals within our organization that currently in France, and working actively with our French counterparts. We are routinely receiving operating experience internal and processing it for applicability on our project, and that's something we're just going to build on.

CHAIRMAN POWERS: You want to be careful on this next one. If you want to live up to his expectation, the next one is an hour and a half.

MEMBER STETKAR: That was just one out of two. You said the next two were going to be two hours.

CHAIRMAN POWERS: That was only one question out of 14.

MEMBER STETKAR: Oh, I see. We're still

on track.

CHAIRMAN POWERS: Oh, okay. Good.

MR. HUGHES: Shall I jump to the second half of this slide?

(Laughter.)

MR. HUGHES: The second half of this slide addresses the panel, and the next slide after expands this. So let's just go to the next slide, if we can, because it's the same as the bottom half of this, only in more detail. But this -- first, let me make an observation before getting into this.

This is very much a program in process. The work that's been done to date has been done at the system level. It is looking at components that everything feeds to system level determination, and clearly we're going to be taking it to the component level and refining the work that's been done.

The Fussell-Vesely and the risk achievement worth kind of criteria that comes in is at the component level. But to get it to the system level, the approach that's been taken is any component touching is enough. The system stays in.

So it's overkill. There will be components that will come out. What we have right now is meant to be bounding encompassing of that information. The second part of this is the --

MEMBER STETKAR: Can I interrupt you there, because I want to talk about the PRA first, because the second part is your expert panel process.

MR. HUGHES: Yes.

MEMBER STETKAR: A couple of statements that I hung up on a bit is the statement that you've concluded that the design PRA model can be used without modification as the plant-specific PRA, because it bounds the plant-specific risk. I was curious about that, because Calvert Cliffs has made a number of changes.

We're not fully informed about the actual plant-specific changes, because unfortunately we've only seen Chapter 8 of the FSARs. It's only the electric power sense. But I'll speak for electric power. There have been changes made to the electric power system, both you know, the site-specific off site power configuration, switchyard configuration and changes made to the on-site electric power distribution system to support things like the ultimate heat sink make up supplies, different numbers of cooling tower fans, non-safety related, but still things that could be in the PRA.

I was curious how you can draw the

conclusion that your generic PRA is a numerical bound for a different configuration, especially in areas that are traditionally rather important to risk, those being power supplies and support systems and ultimate heat sinks and those sort of things?

So I was curious what type of evaluation you actually perform to make that judgment, that indeed we could use the results from the design PRA and be assured that they were conservatively bounding when you're developing component-specific now, numerical importance measures?

MR. HUGHES: Well, we were just caucusing to say, you know, if you want to know exactly what AREVA considered, could we please ask them.

MEMBER STETKAR: By the way, did AREVA make that determination, or did you as the COL applicant, make the determination, that the design PRA was bounding for your plant?

MR. SZOCH: Yes, AREVA made that determination with our oversight and concurrence.

MEMBER STETKAR: Okay. You didn't independently evaluate the PRA with knowledge of your design?

MR. HUGHES: No.

MR. SZOCH: No, that's correct.

MR. REINERT: This is Joshua Reinert from AREVA. I'm the PRA representative. I've been responsible for most of the COLA work to date. One of the first things that I had to do for the Calvert Cliffs COLA was look at any of the site-specific systems, look at the assumptions that we had made for design certification, and compare those with any information that I had specific to Calvert Cliffs.

And then also look at any of the portions of PRA that were not part of the design certification. So just relevant to what we're talking about here, is that first part of looking at the DC assumptions and then making sure that that was bounding for the Calvert Cliffs.

I think your example of the off-site power frequency is just a good example of the kind of -- and I think the answer represents a lot of what we did. For off-site power in design certification, of course we don't know at that point where the site is going to be or really what the switchyard design is, so we use a lot of generic data.

This NUREG, I think it's 6890, provides good generic data. Okay. So that's the DC. Now when I come to Calvert Cliffs, I go back to that same document, and it has historical information for

Calvert Cliffs 1 and 2, which is probably the best information that I'm going to be able to get to use for Calvert Cliffs 3.

I compare that with what we used for the DC, and in that case it was bounding. So although I'm not taking into account specific switchyard design or differences between Calvert Cliffs 1 and 2 and potentially with Calvert Cliffs 3, I'm using information that I think is applicable and generic, in showing that what we assume for DC is bounding.

MEMBER STETKAR: I can understand that for things like the frequency of loss of offsite power. I think what I'm more concerned about is the plant-specific configuration, for example, of the electrical buses. They've changed. Motor control centers, they've added 6.9 kV buses for the ultimate heat sink make up system.

They've introduced an ultimate heat sink make up system. They've switched around the number of cooling tower fans, things like that, that are less obvious to me that simply comparing a generic loss of off-site power frequency, and confirming that that might be bounding for the particular site, given the site's operating history.

So I'm -- you answered one question in

terms of generic off-site power frequency. You didn't answer the question about the configuration of the actual PRA models, and whether that might affect any either qualitative insights or numerical importance measures that may come out from the PRA with respect to either the electric power system or, you know, their now plant-specific design of the ultimate heat sink and the cooling tower fans and things like that.

DC power supplies, they've added some DC power supplies for switching on 6.9 kV buses and so forth.

MR. HUGHES: I think the straightforward answer is to acknowledge that you have a very good point, and to only respond by saying the detailed plant-specific PRA is planned, and we will be embarking on it in the not-too-distant future.

But to date, it has been based upon this type of qualitative assessment, and the PRA for the design certification includes rather large nodes that represent things, and the systems were looked at to some degree to conclude that it was acceptable to consider it bounding.

But the detailed plant-specific calculation has not been done.

MEMBER STETKAR: Thanks.

MR. REINERT: Can I just add one thing?

MEMBER STETKAR: Sure.

MR. REINERT: I just wanted to say that just for the record, I looked at the list of the departures from the latest COLA FSAR, and I've looked at all the site-specific information. I would have to possibly review what I did for specific examples, but I did look at all of those things, the site-specific information and the departures, to make a determination that the DC PRA was bounding and could be used for Calvert Cliffs.

MEMBER STETKAR: Good. That helps. You have to -- also, we're at a bit of a disadvantage in our subcommittee here, because we're receiving both the PRA and the RAP part of Chapter 17, a bit out of sequence from our best, most efficient kind of knowledge process.

Because as I've mentioned, we've only seen Chapter 8 of the COL FSAR. So although we're nominally familiar with the changes that were done in the electric power systems, we don't really know much about any of the other plant-specific changes that may or may not have been done relative to the DCD.

In fact, even in the DCD space, we haven't gone through most of the real plant design in terms of systems. So some of these questions come out in terms of our not knowing what differences may be, but knowing at least in electric power that there are differences, concerns about how they're treated.

You mentioned that the plant-specific PRA, that it will be done some time; I forgot the precise words that you used. When in the grand scheme of populating the RAP list for the COL license will that be done? Will it be done in time to feed into the final RAP list for the COL?

MR. HUGHES: Well, let me respond in a sort of round-about way. Let me go to the PRA first.

MEMBER STETKAR: Okay.

MR. HUGHES: You know, the PRA is an interesting thing, and a lot of discussion has occurred about what kind of ITAAC should be associated with the PRA, and it has to meet requirements when you load fuel. There's language about the PRA being available for audit, for side audit.

The PRA is a living thing, and the PRA doesn't come to a point that it's finished. So the information at the DC stage or at the COL stage is the information available, which is intended to be sufficient to meet the expectations presented.

But refinements will occur. This

reliability assurance panel output will also evolve, and so the process can be clearly agreed upon and communicated and committed to. A list can be provided consistent with that. But I think the expectation should be that there will be modifications to that post-COL.

Because additional design changes or refinements or things may happen in that stage, that will bring it back to the panel, to make sure that it's up to date as the design is refined going forward, leading to the final prior to fuel load list that would be consistent with the full scope, full scale PRA that was intended to support fuel load.

So I think the answer is we will have greater detail. We indicated a few minutes ago we expect to have component level detail in some degree, and the next year to 18 months prior to reaching the point of the COL. But I don't think it's clear that there will not be subsequent changes after that.

MEMBER STETKAR: That's an appropriate around-about answer.

MR. HUGHES: It is a process.

MEMBER STETKAR: Oh, obviously, obviously. It is a process. I'm trying to get my hands around what it is that is produced, in terms of the RAP list at the COL stage, and how that list is then used going forward, both from your perspective as the applicant, because you need to design now plant-specific programs that ensure the reliability, maintenance and reliability of the equipment in this list.

You don't want to be -- I'm assuming you don't want to be in a situation where that's in a large state of flux, where you're not sure what programs you're designing, because you're not sure the scope of the equipment. And from the regulatory perspective, in terms of if the list at the issuance of the COL is simply a snapshot in time, how is that process interpreted from the staff's perspective.

Obviously, you don't answer that. I can ask the staff about that. But I'm telegraphing --

MR. HUGHES: No. I would like to try to respond.

MEMBER STETKAR: Yes.

MR. HUGHES: There is a deterministic approach to components being identified as safety grade or safety-related. This first bullet, and I'm not trying to jump ahead here, but the first bullet underneath the second arrow says is the function used to mitigate accidents or transients.

If a component is used to mitigate an

accident, it doesn't get graded anything. It gets full treatment, and it gets put into the program not as an RAP this or that graded, but it gets everything.

Here, what we're talking about are the components that may be useable or might be used to mitigate, but which are really not part of that original safety grade envelope that is the footprint that assures safety in the days of old.

In the days going forward, we're trying to get those things that might be a little gray, and get them into the program and make sure they go into the maintenance rule, with all the categorization that goes with it. So I think we have high assurance that the footprint, a full footprint should be fine and clearly identified, and these components should be at a high level of confidence process-wise, and there will be a set in there that should be fairly immune to substantial change.

MEMBER STETKAR: Let me ask you about that, since you brought up the list. In the FSAR now, there are three tables, 17.4-1, 17.4-2 and 17.4-3. 17.4-1 is a 14-page table that contains individual components and failure modes.

MR. HUGHES: Yes.

MEMBER STETKAR: And a basis from the PRA for their selection, of why they appear on this list. 17.4-2 is a list of systems that are -- the majority of which are identified by the PRA as candidates for inclusion in the RAP, supplemented by a number of systems that were selected by the expert panel. So that's a system level list.

Then 17.4-3 is additional site-specific systems that have been added to the list, I'm assuming as the result of the expert panel. We'll get into the expert panel in a second here.

My question is, and this is I need some education basically. The final RAP list, as it's implemented in the plant, once you're finally operating, will that list be a list of individual components and failure modes, or is that a list of systems --

MR. HUGHES: Components and failure modes.

MEMBER STETKAR: Individual components and failure modes.

MR. HUGHES: Yes.

MEMBER STETKAR: So that within a particular system, let's say that system has ten components, perhaps only three components within that system may be on your final RAP list?

MR. HUGHES: That's correct. The reason

it is little confusing --

MEMBER STETKAR: It's confusing right now.

MR. HUGHES: Because we submitted a component-specific list, and it would be easy to conclude that that is the list, but it's not. The component-specific list was used to identify the systems to be included, because even within those systems that the component-specific list comes from, there are still design refinements going forward, and we know that list will change.

So the decision was made to provide that information to the staff, and put it in the document, but the decision that controls where we are today is which systems are included, and there will be a refinement of that coming up.

MEMBER STETKAR: I think I'm -- thanks for that clarification, but I think I'm actually now more confused than I was before. If I take the components, from what I heard you say, and make sure that I understand this, that you supplied the component list as evidence to the staff of the process that was used to identify the systems that are listed in the second table, as being populated from the PRA. Is that --

MR. HUGHES: Yes.

MEMBER STETKAR: Okay. Does that mean

that this component list is completely subject to change in the future, or I'm now trying to understand how these three different lists relate to one another. At the COL stage, we would issue a COL tomorrow miraculously. I now have three lists.

If the component list is not complete yet, but the system list is complete, does that mean that Calvert Cliffs will develop Reliability Assurance Program procedures to ensure the reliability of all equipment in each of these systems?

MR. HUGHES: No.

MEMBER STETKAR: Okay.

MR. HUGHES: The intent and the sufficiency of that list of systems is to demonstrate which systems will be in the program, and we're communicating the process, which is the reliability assurance process. But to take and apply some graded requirements to every component within those systems would be excessive. We know that as a bounding list --

(Simultaneous discussion.)

MEMBER STETKAR: Which list is a bounding list?

MR. HUGHES: The system list.

MEMBER STETKAR: The system list is a

bounding list.

MR. HUGHES: Yes. I can reverse the process. The thing that you and the staff and we should be concerned about is to make certain that the systems that are excluded are in fact appropriate to be excluded. That we've been concerned about, and that the staff has been concerned about, and that we have responded to RAIs to address.

MEMBER STETKAR: Gene, I'm glad you said that, because that indeed is, you know, you've checked off the box on one of my questions here. Because it's very, very important at this stage in the game to have very good confidence that there is good reason that a system is not on this list, if indeed the system level list is the only determining factor at this stage of the licensing process. So I'm glad you mentioned that.

MR. HUGHES: I would also like to characterize. It would be inappropriate for me to communicate that the component information is worthless. It's not. The component information is information derived from the existing PRA, which we have confidence in.

But it would be equally wrong to communicate that it will not evolve between now and the license being issued. It will. MEMBER STETKAR:

MEMBER STETKAR: What I wanted to ask you, just to make sure that I understand a bit of the process, it's certainly not appropriate for the ACRS to be reading lists of components and trying to second-guess which one is on and which one isn't. That being said, I tried to do that.

MR. HUGHES: I did read the transcript of a prior meeting where you received the list, and noted that you had some excitement and concluded you might look at it.

MEMBER STETKAR: Well, I didn't read the whole thing, but I didn't have to look far. Just out of curiosity, because of an -- I want to understand the process and have confidence that the process will eventually develop a list at that component level, so that indeed we can have assurance that the list is reasonably complete, that it's just well-justified, and that indeed the plant will develop site-specific programs to ensure that the reliability of equipment and systems are maintained.

One thing I noticed on this list, which was derived from the current PRA, is that there are no manual valves on this list. You say well, how can I maintain a manual valve? Well not necessarily the valve itself, but you can do an awful lot of testing to make sure that indeed the flow path is open.

There are spurious actuations of specific components, like motor-operated valves, if the valve fails to stay open or closes spuriously, that are identified as with high risk achievement worth from the existing PRA. I think one in -- come from a clean water supply. It's actually a return valve from I think some chillers. I'd have to go look at my list here.

So I went, and I looked at the, you know, P&ID for that system. I noted there are two manual valves in series on the supply side and the return side, that if those close spuriously you would block flow. Now Fussell-Vesely importance isn't going to be very high for those, because the failure rate is pretty low.

But I would expect the risk achievement worth to be the same as that motor-operated valve. They're not there. The question is are they not there because of the process that was used to populate this list, or are they not there because the valves are not in the PRA?

MR. REINERT: Joshua Reinert from AREVA

PRA. I have to look at the model to verify for sure, but I can logically conclude that those manual valves are just not modeled in the PRA.

MEMBER STETKAR: Okay. I'm assuming that when you do your site-specific PRA, you might --

MR. HUGHES: I would rather give you assurance that they will be included, so you don't have to assume.

MEMBER STETKAR: Again, you know, we're kind of stuck between switching pages of detail and yet questions about the tools that are being used to generate this detail, versus some future tool, and to develop assurance that the process will have the tools in place in a manner to support what's necessary for at least the COL license.

MR. HUGHES: I readily acknowledge that what's provided there is an intent to provide an example, and it's insufficient to meet the task.

MEMBER STETKAR: Okay, thanks. That -- thanks.

MEMBER RAY: John, could I ask a question?

MEMBER STETKAR: Yeah, sure.

MEMBER RAY: You said it might, there would be changes, I think you said Gene and emphasized, through the time the license is issued.

You didn't mean to say then there would be no changes after that?

MR. HUGHES: They would certainly occur prior to the license being issued, but there will be other changes after that. Yes, thank you, Harold.

MEMBER STETKAR: So I've still got a few minutes here. I'm trying to get through -- this is just a minor nit, but again on the process, and it feeds into what Harold just brought up. You said that after the, after the license is issued, there's a -- the PRA models will be updated at least once every three years, every 36 months.

There was a statement that says that the data will be updated once every four years. Why are those different?

MR. HUGHES: I think we're going to have to dig into that.

MR. GIBSON: I don't know offhand.

MR. HUGHES: I have no idea.

MEMBER STETKAR: I mean there is a commitment. Obviously, there's a commitment to keep the PRA up to date and updated periodically. I was just curious whether that was a conscious -- it's in the same paragraph. It just sort of leaped out at me.

MR. GIBSON: Do you have a reference right there?

MEMBER STETKAR: It's in Section, if you're ready, 17.4.4.1.1.4 of the FSAR.

MR. GIBSON: You beat me. I couldn't get --

MR. HUGHES: I have it. I have it, yeah.

MR. GIBSON: Do you have it?

MR. HUGHES: Yeah.

MEMBER STETKAR: But I was just curious whether it was a conscious decision, or whether there was some other rationale. It's not really pertinent to the discussion we're having today. I was just asking.

And now I have a couple of other questions. We'll get to the expert panel, but I want -- these are all detailed sort of body count type questions. If you want the reference, in Section 17.4.4.1.2.1 of FSAR, you discuss the development of the master equipment database.

I guess now I understand that is the vehicle by which you will actually implement the RAP; is that correct? In other words, that will be the official database of equipment.

MR. HUGHES: Yes.

MEMBER STETKAR: Okay, okay. Now I

understand that. Some things again about the process. In those sections of the FSAR where you discuss the development of the database and the use of that database and feedback through design change process and things like that, there are constant references made to reevaluating the effects of design changes on the risk-significant components that are in the database.

From what I had to deal with, that was this 14 page list of equipment in Table 17.4-1. My question is shouldn't we be looking at not only the effects on the things we've identified, but reevaluating all of the things that are not on that list, to see whether they might become important?

MR. HUGHES: Yes.

MEMBER STETKAR: That sense wasn't in there.

MR. HUGHES: Okay.

MEMBER STETKAR: Because it constantly said well, we'll take a look at the -- we have a list now. We'll take a look at the list and see whether our insights about anything on that list changes.

MR. HUGHES: No. Your belief is consistent with our intent, and we can fix those words. Absolutely.

MEMBER STETKAR: Okay, okay.

MR. HUGHES: It is an unbiased look.

MEMBER STETKAR: Okay. That's all. Now you can start talking about the second bullet.

MR. HUGHES: The one thing I want to point out, in order to put the second bullet into clarity, is what we have in the FSAR is a series of attributes that are identified with weight and with frequency and with impact definitions, and when you talk about deterministic risk ranking, and I want to read you one sentence that's easy to miss in the text, but it's there for a reason.

It says "Although some of these definitions are quantitative, both of these sets of definitions referring to the deterministic, are applied based on collective judgment and experience."

I want to point out that the numerical ranking that's in the FSAR is intended to be applied as we get to the component level, and these factors were considered in the systems. But the judgment was made based more on judgment than on this kind of numerical analysis in the work that's gone on to date, with the error being to include a system rather than exclude it.

So these are the types of issues that are raised. They're shown on this chart, and you just simply look at them. One thing that is not on this list is is the function or the component already safety grade, already included in the footprint. If that's the case, it comes in. There's no change to that footprint.

Now when we get to a particular component and we determine subsequently that it may not be as critical to the operation, then we might consider to change that. My diesel generator is a perfect example. It comes in originally as a safety grade component, but then it can get refined, as any system, any set of components can. Are there questions before I --

I'm sensitized to the time to cover these points.

MEMBER STETKAR: We're still ahead right now.

CHAIRMAN POWERS: Oh, we're fine.

MR. HUGHES: Oh no. I didn't mean to curtail you. I meant I didn't want to go too fast.

MEMBER STETKAR: No, that's fine. I'm perfectly happy to stop you now. When you talk about components, you're talking not only about components, but in the final list components and failure modes. Is that correct?

MR. HUGHES: Yes, yes.

MEMBER STETKAR: So that, for example, a motor operated valve can have, you know, in principle four failure modes, fail to open, fail to close, spuriously open, spuriously close.

If the process identifies only one of those particular failure modes for that valve that's risk-significant, then your Reliability Assurance Program will be focused solely on that one failure mode for that component; is that correct?

MR. HUGHES: That's correct.

MEMBER STETKAR: This process, by the way, I have not done any work at all in the maintenance rule arena, so I'm not completely clueless but mostly clueless about the maintenance rule. Is this ranking pross -- or I don't want to call it ranking process, selection process that you've applied the same as that used for operating plants, in terms of populating equipment to be tracked by the maintenance rule program?

MR. SZOCH: Yes, very similar. Same similar approach that we've used on existing maintenance rule panels.

MEMBER STETKAR: When you say "similar," is it in terms of the numerical rankings and the way that this formalism is documented in the FSAR, your FSAR?

MR. SZOCH: This level of details is not -- in the operating plants, is not in the FSAR with regard to the specific deterministic approach. But our procedures outline, of course, PRA being a major input. But the general deterministic approach, maybe not quite as objective with the ranking. But the questioning and the use of this type of information to make the determination of in scope or out of scope for maintenance rules is what we currently use.

MEMBER STETKAR: What I was getting to Rick is because I don't have any experience with the maintenance rule, was the -- you characterized it as objective numerical, what I'd call weights for each of those five different questions that you have on the screen now, and then weightings in terms of the quality from 1 to 5 for the answers to each of those questions.

That is more quantitative than the typical maintenance rule process?

MR. SZOCH: Yes. That is, yes. It's not quite as precise, but there is, I'd say, a relatively objective approach, and also a little bit slightly more scientific than that.

MEMBER STETKAR: Has this particular numerical process been used anywhere, or is this something that was developed specifically for this application?

MR. SZOCH: This can -- can you help me with that?

MR. HUGHES: I can respond to that. This approach, I believe, is consistent with the 10 C.F.R. 50.69 type of implementation of the South Texas project.

MEMBER STETKAR: Is it? Okay.

MR. HUGHES: I believe it's consistent. I can't say that it is the same.

MEMBER STETKAR: As I said, I don't know. That would help an awful lot in terms of trying to understand whether this is, you know, a novel approach that what the staff needs to review or think about in the context of specific numbers and implications, or whether this is something that indeed has been used in the past.

MR. SZOCH: No. This is, as Gene said, modeled primarily off the South Texas procedures.

MEMBER STETKAR: Thank you. Go on. I'm just writing things down.

MR. HUGHES: Okay. Next slide. Jumping

ahead to 31, there are some questions that have been raised by the staff, and recently there has been response provided to them over the past two or three months.

The first one deals with reliability and availability assumptions translated into verifiable attributes. The point I would make here is that a PRA assumption that a component has a failure rate of .0015 is not the type of verifiable attribute that would fit into the program.

Rather, it would be a sense of the function and its capability and its general reliability, and then we would apply things to it. Systems included has looked at the systems the staff identified and responded, that we would expand the list to include additional systems.

The system boundary has been explained, and a recent response in terms of the types of supporting components that are included within a system, and the criteria for selecting the panel and the rationale has been addressed.

But this is in-process all information that's been provided to the staff very recently. So it's probably premature for the staff to be able to comment much beyond they've gotten it.

MEMBER STETKAR: Gene, the system boundary, you said the question there, in principle we have access to all of the RAIs, and in practice if we ask for all of them, we get them, which is neither fair to the staff nor us.

CHAIRMAN POWERS: If you ask for one of them, you get them all.

MEMBER STETKAR: Well no. That's not necessarily true. But the system boundary question was the boundary of the interfaces between the system and its support functions, things like cooling water valves through a heat exchange or --

MR. HUGHES: That's correct.

MEMBER STETKAR: Okay.

MR. HUGHES: Yes, that's correct and that's what we responded.

MEMBER STETKAR: Because I was curious again from the specific list. I was trying to understand what that might meant. For example, the CVCS, parentheses (including reactor coolant pump seal injection), is listed as one of your systems, and yet again, back to the component list, there are indeed only two motor-operated valves on the whole system, and they're the low pressure letdown line.

So I wasn't sure whether the boundary

taking bits and pieces of the piping of particular systems, or whether it was more of the boundary of the interface with the so-called front line and support systems, and it's the latter.

MR. HUGHES: Yes.

MEMBER STETKAR: Okay. Parsing up the system among its individual branch lines and valves has yet to be done.

MR. HUGHES: Making sure that nothing is missed, because it was somehow on a bias one way or another for a cut.

MEMBER STETKAR: But right now, since CVCS, including reactor coolant pump seal injection is on your list, in principle, every filter and demineralizer and everything in that system is on the table right at the moment.

MR. HUGHES: That's correct.

MEMBER STETKAR: Okay, thanks.

MR. HUGHES: Last slide is just the additional refinement of the most recent question, and again, you'll see from the date this information was just submitted in the last few days. Okay.

To sum up on the Reliability Assurance Program, I think we've spent a fair amount of time talking about it. I think it's a good program.

It's one that provides a mechanism to transfer this information to the plant and to the various functions that support the construction and testing leading to the plant being able to carry it forward, and it's a system that I think assures a high degree of confidence once implemented, that these things will be done in a proper fashion. So thank you.

CHAIRMAN POWERS: Are there additional questions on this subject on this particular presentation? We have more to go here. I think we can just charge right ahead.

MR. HUGHES: Okay. Next slide.

MR. SZOCH: Shall I move to the maintenance rule?

CHAIRMAN POWERS: Yes.

MR. SZOCH: Page 34, let's go right to 35.

CHAIRMAN POWERS: I'm pointing out to him, Mr. Quality Assurance here, that he told me an hour and it was 55 minutes.

MEMBER STETKAR: Okay.

CHAIRMAN POWERS: He's just not -- the man just cannot speak with precision. That's all there is to it.

MEMBER STETKAR: I'm lucky I can speak.

(Laughter.)

CHAIRMAN POWERS: It was within the uncertainty band.

MEMBER STETKAR: Plus or minus --

(Simultaneous discussion.)

CHAIRMAN POWERS: That's the last recourse of a scoundrel. It's in the uncertainty.

MR. SZOCH: Okay. With regards to maintenance rule, we have on page 36 and 37 nine COL information items on maintenance rule. They're all being addressed through our incorporation of NEI 07-02 Alpha, which is the generic template.

Actually, we outline that template on page 21. We're talking about the maintenance rule a little bit. But the key elements as we showed, really a key part of the process in the operational phase, of course on page 21.

Of course, we'll be implementing NEI 07-02 Alpha verbatim into our program and into our process, just as we have in the operating plants. So we don't intend to deviate from that at all.

As far as the -- on page 39, the SER open item. Actually, the top RAI 192, it's actually encompassed by that same issue. That's the use of industry experience, operating experience. That's per

NEI 07-02 Alpha we'll be doing that.

And then RAI 228 was recently responded to, where we intend to implement readiness milestones as the license conditions for the maintenance rule program. So we intend to do that. And on page 40, there's one SER confirmatory item. That's responded to via RAI 62.

That again is to outline our intent to use industry experience as part of our maintenance rule program, which of course we'll do. That concludes the section on maintenance rule, unless you wanted to go into more detail there. We did touch on it earlier.

CHAIRMAN POWERS: I mean the question is not different than what we discussed before. It is you have kind of an AREVA-centric thing right now with the design coming down. Sooner or later, you're going to put your imprimatur on with your own corporate culture.

MR. SZOCH: Yes.

CHAIRMAN POWERS: It's going to be very interesting to see what changes, what evolutions occur there. I don't think, you know, I don't think either organization is static, and it's going to be kind of interesting to see how much AREVA affects Calvert Cliffs, and how much Calvert Cliffs affects AREVA here.

MR. SZOCH: With --, that's a future thing. We've actually already started pieces of maintenance rule today, as a matter of fact. We have done the initial phases of scoping, recognizing that using that PRA list, John, that you had in your hand there, that is an initial input as to what will be within the maintenance rule.

So we're already flagging that equipment and going through that process of teamed approach between UniStar and AREVA. So we're doing that together, and Constellation and the industry has a lot of maintenance rule experience, and we're sharing all the procedures and processes with AREVA.

So we're actually injecting that culture in the way we address maintenance rule today in the design process.

CHAIRMAN POWERS: It would be fun to be fly on the wall for both the discussions together and the discussions subsequently --. Okay. Well thank you very much. Surinder, I may turn to you for some advice. Should we go ahead with the staff presentation on this?

MR. ARORA: We have the staff presence here, so we can do that.

CHAIRMAN POWERS: Okay.

MR. ARORA: Can we start?

CHAIRMAN POWERS: Yes. We can just wrap this one up. I think it's fair to say we understand pretty well how the COLs and open items are going to be addressed, and so we're looking mostly for insight from the staff.

Thank you very much. This was a very, very edifying discussion, both philosophically and specifically. It was well worth my time to participate, and I thank you for your willingness to go along with our philosophically and speculative questions.

MR. GIBSON: Thank you.

CHAIRMAN POWERS: We'll look forward to -- maybe we can have you guys back to give us an update as this progresses on, because it will be very interesting.

MEMBER STETKAR: Who is the PRA D&A?

(Pause.)

MR. ARORA: Are we ready?

CHAIRMAN POWERS: We're ready.

MR. ARORA: Let me introduce Tarun Roy. He happens to be the chapter PM for --

CHAIRMAN POWERS: This man is seriously

busy. You're overworking this guy.

MR. ARORA: He wants to be done with this chapter.

(Laughter.)

CHAIRMAN POWERS: I hate to tell you this, but I just told this guy we wanted him to come back and talk to us some more. That means you have to come back and talk to us some more.

MEMBER STETKAR: Reorganize your retirement plan.

(Laughter.)

MEMBER SHACK: Accelerate it.

CHAIRMAN POWERS: It's not the individual; it's the institution.

MR. ROY: Anyway, I'm back again. We have NRC technical staff involved with the review of Calvert Cliffs FSAR Chapter 17-R, Hanh Phan, and we have also technical staff, Jonathan Luciano-Ortega. He did work, and his branch chief is there to support us, Mr. Juan Peralta.

CHAIRMAN POWERS: And we know why. Jonathan --

MR. ROY: During this meeting, the staff plans to make a presentation of the Chapter 17 quality assurance and reliability assurance safety evaluation report with the open items. I'll go back to the next slide. This is overview of the staff review. 17.1 is the quality assurance during design. This is a complete IBR.

17.2, which is the quality assurance during the operational phase, which is a QA program, is provided in Section 17.5. 17.3, quality assurance program description, this is IBR. 17.4, reliability assurance and 17.5, quality assurance program description, 17.6 is description of applicant's program, and 17.7, maintenance rule program.

We have a number of RAI questions, 15. Out of that, nine are from Reliability Assurance Program and one from 17.5. I am representing 17.5. There is only one open item in this one. The staff issued RAI 200, Question 17.5.6, to request that the applicant commit to following the guidance in Reg Guide 1.33, Quality Assurance Requirement and 1.33 Operation, and issue a Revision 1 to QA PD UNTR 06001A accordingly.

We're waiting for that response. That is in preparation from UniStar.

I would go back to now Hanh Phan, 17.04. If you have a question on 17.05 of the quality assurance programs. Otherwise, we'll go back to

17.04.

CHAIRMAN POWERS: Any questions on that one? I guess not.

MR. ROY: Hanh?

MR. PHAN: Thank you, Tarun. Gentlemen, good morning -- actually good afternoon by now. My name is Hanh Phan. I am back. Two months ago I presented you the staff evaluation of U.S. EPR FSAR Chapter 17 and 19, and today I'm going to present you the staff reviews of the Calvert Cliffs FSAR, Section 17.4 and 17.6.

I would like to start my presentation with a brief introduction of myself again. I joined the NRC in 2006. Prior to that I work at the National Labs at the nuclear power plants. I have over 20 years' experience in nuclear, specializing in reliability and PRA.

I earned the Bachelor and the Master in Electrical Engineering at the Washington State University. With that, I would go straight to the open items in SECTION 17.4, reliability assurance programs. At the end of Phase 2, the staff identified seven open items.

The first two are recently responded by the applicant. The staff intentionally keep Question

17.4.3 open, because we are finalizing the interim staff guidance. In the response to this question, the applicant agrees to revise the FSAR to be consistent with the staff interpretive guidance.

So with that, because we have not finalized the guidance, so that we have to keep this question open.

MEMBER STETKAR: Hanh?

MR. PHAN: Yes sir.

MEMBER STETKAR: What's the schedule for that, the ISG?

MR. PHAN: Before answering your question, may I introduce Dr. Todd Hilsmeier. He's the Office of the Interim Staff Guidance, and actually, we planned to issue that last month. Just because we -- the question with the working group, ISG working group, and they provide us more comments.

Even though we issue a call for public comments, now we're receiving more comments. So we are planning to if possible to issue the final this month, at latest next month.

MR. HILSMEIER: Yeah. My name is Todd Hilsmeier, and we've been working on the ISG for the past year, and the ISG is ready to go, except for the D-RAP ITAAC. We're finalizing wording in the ITAAC for D-RAP.

When we, back in October, we sent out the ISG for public comment. We had a few editorial comments that we incorporated. Since then, there is a little more feedback from the industry on our proposed D-RAP ITAAC. So we're currently finalizing that, interacting with the industry also.

We're hoping within the next few weeks we can finalize the D-RAP ITAAC, so the ISG can be published.

CHAIRMAN POWERS: It's imminent in the sense of the next month or so?

MR. HILSMEIER: I'm hoping so very much.

CHAIRMAN POWERS: Are the -- just out of curiosity, I've seen the draft. Are the -- do you anticipate substantive changes in the final?

MR. HILSMEIER: No. The public comments were editorial in nature, three of them. So we incorporated all of them, and the additional changes are editorial, just to make sure the ISG is clear. We don't want to miscommunicate to the public our expectations of RAP.

But as far as technical changes, there isn't, other that maybe a D-RAP ITAAC, which we're working on finalizing the words.

CHAIRMAN POWERS: Trying to get more specificity in them?

MR. HILSMEIER: Yeah. The overall meaning's the same for D-RAP ITAAC. But the wording may change slightly. Thank you.

MR. PHAN: For the next question, 17.4-4, in the response to this question, the applicant agreed to include these systems, including the fire water distribution system, the sprinkler system, the spray deluge system and core melt stabilization system in the scope of the D-RAP.

The staff kept this question open, because the applicant referenced a wrong response from the AREVA response to the staff RAI during the DC review. Most of the responses to these questions are -- they met the staff expectations. So the staff considers these questions will be resolved soon after today's meeting.

For the next five questions, most of the questions was issued recently as a result of the previous meetings, the ACRS Committee meetings on the AREVA DC Chapter 17. The five questions on Question 5, that's on the system boundaries.

Question 6 on the rationale for the criteria used for selecting the expert panel, Question

7 on the rationale used for the deterministic process, Question 8 on the performance criterias and goals for the risk-significant SSC identified by the deterministic process, and the last question, Question 9, on the inclusion of the normal heat sink, start-up and shutdown system, the aux cooling water system, the closed cooling water system and raw water supply system from the scope of the D-RAP.

The applicant has not responded to these questions, so the staff is going to -- the question as open item.

CHAIRMAN POWERS: We discussed criteria for selecting the expert panel at some length in the previous hour, 55 minutes, I'm sorry, and what we had learned is that the expert panel currently is an AREVA-centric operation and that it will eventually transition into a more site-specific.

So we may ask for the rationale for the selection. What are you looking for? I mean how did they give you an adequate response?

MR. PHAN: Please give me one second. In FSAR, Section 17.4.4.1.3, expert panel, the applicant cited that for the expert panel as a minimum combined of expert panel working -- at least three individuals with a minimum of five years' experience. For those criteria they use to select the expert panels.

The staff would like to ask for the basis for selecting or the assignments that three years be -- five years, that that would be appropriate, and the basis behind those numbers.

CHAIRMAN POWERS: I'm still trying to understand what an adequate response is. They're going to come back to you and say well, everybody that we selected on this has 25 years' worth of experience.

I mean does that meet your expectations, or do you ask the question Commissioner Apostolakis will ask if he is confronted with this question, and he says "How do you know this guy hasn't been wrong for 24 of those 25 years?"

MR. PHAN: The answer is if the response says the expert panels would have more than five years' experience, then the staff will accept it.

CHAIRMAN POWERS: So they're meeting your criteria and this is really what you're looking for.

MR. PHAN: Yes, yes.

CHAIRMAN POWERS: Okay. Well, that's fine. I understand. But do you worry at all about this transition from a, what I call an AREVA-centric expert panel that one that's more site-specific?

MR. PHAN: Yes, and the meantime, the

staff have no detailed information. However, in their FSAR, the applicant's FSAR, Section 17.4.4.7, Records, say that the records of the expert panel decision and supporting documents are retained as QA records.

If the panel decision and meeting minutes included actions and resolutions, so the staff would have the opportunity to look at these documents at the inspection, prior to the closing of the ITAAC.

CHAIRMAN POWERS: There's no question that you have access to all the information you need on this rather critical operation. I guess I'm trying to understand, is this something that I should be tossing and turning at night over, or is it something that you think is just a walk in the park, it's going to happen, and all we have to do is make sure it does?

MR. PHAN: I lost your point, sir.

(Laughter.)

CHAIRMAN POWERS: Is it something to worry about, or is it something I should let water roll off the duck's back here?

MR. PHAN: Yes please.

(Laughter.)

CHAIRMAN POWERS: Okay. I don't have to worry about this.

MR. ARORA: Before you go further, I just want to put it on the record that UniStar has responded to RAI 224, and that was dated April 16th. However, staff hasn't had time to look at it.

CHAIRMAN POWERS: Thank you.

MR. HILSMEIER: If I may say a few words.

CHAIRMAN POWERS: You'll have to tell us who you are.

MR. HILSMEIER: Todd Hilsmeier.

CHAIRMAN POWERS: And you have to tell us something about yourself. See, you don't get to sit up there for free. We impose a certain penalty.

MR. HILSMEIER: That's not too bad of a penalty. My name is Todd Hilsmeier, and I have a Master's and Ph.D. in Nuclear Engineering, and I work at a consulting company doing PRA for five years.

CHAIRMAN POWERS: We don't hold that against you.

MR. HILSMEIER: Okay, good. It's a different experience. Then tired of traveling, because PRA work back then was drying up.

CHAIRMAN POWERS: You get no sympathy from this panel, by the way.

(Laughter.)

MR. HILSMEIER: After my consulting days, I joined Salem and Hope Creek nuclear power plants, where I performed risk assessments for about six and a half years. Then I joined Diablo Canyon and performed PRA and risk analyses for another two years.

Then the job opportunity came available to work with NRC, and I've been performing, been a reliability risk analyst at NRC for about five and a half years, and I'm here today. Now I forget, though, what I was going to talk about.

(Laughter.)

MR. HILSMEIER: The FSAR, the main purpose of 17.4 in the FSAR is to describe the Reliability Assurance Program that will be implemented during the design and construction phases, and also how it will be integrated through an operations phase.

During the application review, we've reviewed the process. During the design construction phases, we reviewed plans to perform inspections, to make sure that the process is implemented correctly. So we'll be evaluating how the maintenance rule -- not the maintenance rule, how the expert panel, the D-RAP expert panel interacts and conducts their business, and looking at their meeting notices, and making sure they maintain and update the list of risk-significant SSCs.

Those risk-significant SSCs are, I'm going to use John's word during a former COL ACRS meeting. The SSCs are effectively populated during the design certification phase. During the design certification phase, they limit the scope to the design certification, the scope of the design certification.

During the application phase, the COL applicant needs to update the list to include those SSCs from the design certification plus any plant-specific SSCs. At that point, the list should be effectively populated.

However, because the PRA does change over time, since the PRA's a live PRA model, the list may change, but we don't expect it to change much during the design construction phases.

MEMBER STETKAR: Let me stop you there, Todd, and ask what is the staff's expectation in terms of completeness and level of detail of the RAP list, at the time the COL is issued?

MR. HILSMEIER: We expect -- to the answer to that question, we first need to know --

MEMBER STETKAR: Let me ask it in terms of I don't want body count, but we heard from the previous presentation that indeed the Reliability Assurance Program would be implemented finally, looking at individual components and even specific failure modes for those components.

MR. HILSMEIER: Right.

MEMBER STETKAR: Do you expect to have a list at that level of detail at the COL licensing stage?

MR. HILSMEIER: We're not requesting that they have the dominant failure modes identified.

MEMBER STETKAR: Okay. But you do want a list of individual components?

MR. HILSMEIER: Correct, and the reason for that, because they may actually be implementing RAP during the application phase, there's two parts of RAP. The RAP is implemented in two stages. First is design reliability assurance program, which is RAP, implementation of RAP during the design construction phase.

Then the second stage is RAP during the operation phase, which is integrated into existing programs, and I'll focus on D-RAP here. D-RAP essentially has two parts. The first part is applying the essential elements of RAP, such as organization, making sure that organizations are interfacing to ensure the PRA model is consistent with the design constructive plan, and also making sure that the RAP process is proceduralized, and that records are maintained.

And that if there's deficiencies found in the D-RAP program, that they're efficiently and effectively corrected. So that's the corrective action process. So we call, I believe there's five parts to essential elements. Then the second part of D-RAP is making sure the -- subjecting the non-safety related risk-significant SSCs and scope of RAP to the quality assurance controls under Section 17.5.

We don't mention anything about safety-ready SSCs, because they're already subjected to Appendix B. So D-RAP is two parts, applying the central elements of D-RAP and subjecting the non-safety related SSCs that are It can have a risk-significant to quality assurance controls. Those are defined in Section 17.5 of the FSAR.

MEMBER RAY: They aren't from Appendix B though?

MR. HILSMEIER: No. The Part V, the acceptance criteria that the staff reviews, the non-safety related quality assurance controls are in Part V of SRP 17.5. You could think of them as graded quality assurance controls. They're not as stringent as Appendix B.

MEMBER STETKAR: Since you're the PRA guy at the table, how do you answer a question related to the example that I brought up in the previous discussion, where based on a component that I can see in the table, going back to check whether or not other components that have similar failure modes should -- why they are not in the table, and they're not in the table.

I get an answer that well, they need to go check the PRA, because maybe they might not be in the PRA. If indeed the PRA is used as the basis for populating this list, which now includes individual components, forget the failure mode for the moment, but individual components, what type of assurance do you have that indeed the tool that's being used to populate that list is adequately detailed and complete enough to support that function?

MR. HILSMEIER: That's a good question. As you just said, PRA is a tool, and NRC is a risk -- we're risk-informed and not risk-based. So we use other tools to ensure that the list is complete. One of those tools is industry use of operating experience, and also expert panel.

Expert panel has a very important part in contributing to ensuring the risk-significant SSCs are of sufficient quality.

MEMBER STETKAR: Well, but I'm going to hang up on my specific example here, because I have a motor operated valve that is on the list, and I have two manual valves in series with that motor-operated valve that are not on the list.

What type of confidence do I have that the process will work, such that the panel, who has now the complete oversight and full knowledge of every piece of equipment in that plant, will say "Ahh, we need to put those two manual valves on this list, because the PRA didn't identify them as important. But we're so knowledgeable of every single component in the plant that we realize that those two specific valves were omitted."

How do I have confidence that that process will work, if indeed the numerical mechanical tool that I have isn't throwing that on the table in front of me?

MR. PHAN: I would like to answer your question here. As you're aware of the COL applicant's reference and list from the AREVA DC FSAR, it's exactly the same list, 17.4-1 table and 17.4-2. In addition, they go by this list of site-specific systems.

The staff beware of in the PRA some manual valves, especially in series, not included in the PRA. And the staff beware that the PRA may not actually -- the PRA does not include all of the components within the plant. However, because we don't have the PM --, there's no way for the staff to verify any components that not on the list.

MEMBER STETKAR: Excuse me, Hanh.

MR. PHAN: Yes.

MEMBER STETKAR: I didn't make up these two manual valves out of thin air. I got them from drawings that are in the DCD.

MR. PHAN: Yes.

MEMBER STETKAR: You know, I don't have things that you don't have. So --

MR. PHAN: Yes sir. I have more of the final PRA, and I had to go back and includes the manual valves, because that not in the baseline interim event PRAs. So I beware of those kind of scenarios, that the valves or the components not included in the PRA.

But to answer your question, the staff don't have any confidence at this level to say all of the components are in the RAP.

MEMBER STETKAR: But I haven't seen -- it might be hidden in another RAI, but I don't think I've seen any questions regarding the completeness of those lists. I was struggling with the list obviously, as you heard from the previous.

MR. PHAN: Yes.

MEMBER STETKAR: But from what I hear you saying, is that at the COL stage, you expect to have a list that's comparable to Table 17.4-1; is that correct, that lists individual components, not the system level list but individual component list?

I've seen the questions you've asked about the third bullet on the slide that's out there, about excluding, you know, justification for exclusion at a system level. Why isn't a system on that second table. But I haven't really seen any questions about justification for completeness of the list of components.

So I'm concerned that that process, in terms of how do I have confidence that the list is complete, recognizing that the PRA will change in the future.

But I think at the COL stage, I do not like to see the possibility that the list contains, pick a number, 100 components now and that three years from now it's going to contain 500 components, you know, plus or minus a few here and there, as you gain a little bit more experience, you know, as to be expected.

But that the list, if indeed the expectation at the COL is that the list is reasonably complete, how do you ensure that the process has been brought to fruition?

MR. PHAN: Yes. The staff beware that Table 17.4, listing just about 140-some components, and that not all the components within the plants. There are many, many more. The staff relies on the expert panels, the applicant's expert panels to put out the component level.

However, at this point, based on the information and the designs available, they can only go up to the system levels. I beware that in some significant systems, the components are not there like the manual valves you mentioned. They still not in the PRA, and they not saw on the list. The staff --

MEMBER STETKAR: This comes back, though, to Dana's question, is that we heard that they've already convened an expert panel, with nine members with infinite experience.

MR. PHAN: Yes.

MEMBER STETKAR: And indeed they did, that expert panel actually did, in their defense, add a number of systems that for whatever reason were not identified in the PRA. So it's clear that the expert panel indeed was thinking beyond the bounds of the PRA, thinking about other factors. That's good confidence, at least at the system level.

Yet there's no evidence that they added any individual components, or that were given the charter or the guidance that indeed they needed to think about adding any individual components, which is a huge endeavor. You know, I don't know where else to follow this, is just I personally feel a bit uneasy about that part of the process.

If indeed at the COL stage the goal is to have a reasonably complete, I don't want to say 100 percent complete, but reasonably complete component level list, if the goal is that and not just systems.

MR. PHAN: Yes.

MR. HILSMEIER: So in general, is your question how will we know that the process is appropriately implemented?

MEMBER STETKAR: Well, there's the process part of it, but there's also what level of review does the staff do, to ensure that the tool, if I'm using the PRA as the only tool that I have now, aside from the expert panel's individual, you know, experience and expertise.

What level of review does the staff do to ensure that the PRA is sufficient completeness and level of detail to satisfy the requirements to generate this list?

If the PRA is not the fundamental tool to generate the component list, then how do we ensure that the expert panel has sufficient guidance and experience, breadth and depth, to recognize the fact that perhaps they need to identify, you know, a nine year-old valve here and a nine year-old valve there, or a pressure instrument here and a pressure instrument there?

MR. HILSMEIER: Because those things may not have been included in the PRA at that level of detail.

MEMBER STETKAR: That's a much different charter for that expert panel.

MR. HILSMEIER: Right, yes.

MR. PHAN: The staff knows that the PRA shall be developed based on the P&IDs and the final design of the plant. However, the staff does not have any P&IDs.

So we rely on the system description. If the system description does not identify the valves or even the pumps, then that not verified by the staff, that the components correctly models in the PRA or not.

MEMBER STETKAR: Thank you.

MR. HILSMEIER: Well, if the PRA is sufficient for use in the maintenance rule, then it should be sufficient for use in RAP, because the two uses are very similar. Now would the PRA at the COL application phase be sufficient for use of that maintenance rule? That's the question.

MEMBER STETKAR: That is the question.

MR. HILSMEIER: There are reassurances that the risk-significant SSCs is continually updated, and so when that plant-specific period model is developed, what is it, one year before fuel load, in accordance with the standards?

MR. PHAN: At the fuel load.

MR. HILSMEIER: At fuel load, the RAP list would be updated relative to that PRA model, which is also used for maintenance rule.

MEMBER STETKAR: I think what I'm asking, I understand that, is what likelihood do we have that the RAP list might increase by a factor of three or four in magnitude when that PRA is developed at the time of fuel load, and that PRA is not subject to a staff review. It's subject only to audit, because it's past the COL issuance at the time. You see my concern.

MR. PHAN: We understand your concern.

MR. HILSMEIER: I mean maybe this may answer the question, is what do we do with that list of risk-significant SSCs? Because of the list is set on a shelf and we didn't do anything with it, it really doesn't matter what the quality of the PRA is.

So what we do with the list of risk-significant SSCs are three things. I'm going to go last, I'm going to start from the last and go to the beginning. The first thing is the risk-significant SSCs are considered high safety significant in the scope of the maintenance rule, which means they're given explicit specific reliability performance criteria based on the PRA. We expect that to be done when the plant-specific PRA model is developed.

Also, the risk-significant SSCs, we identified dominant failure modes, to ensure that maintenance and testing activities address them. If that's done, if a new SSC is identified under the plant-specific period model that's developed for fuel load, that should be sufficient, because there's being these testing activities that are performed during the operations stage.

And also the risk-significant SSCs are subjected to quality assurance controls during the design construction phase. If it's safety-related, if you list a safety-related SSC it doesn't matter. It's subjected to Appendix B.

If it's a non-safety related SSC which we didn't identify under the COL application PRA model, but it's identified later on, we would ensure that that new non-safety related risk-significant SSC meets the quality assurance controls, that we need to verify that the quality assurance controls are met.

And also, that risk-significant SSC would be subjected to quality assurance controls thereafter.

MEMBER STETKAR: Going forward, but not -- it's already installed in the plant at the time of fuel load.

MR. HILSMEIER: Right. Malcolm, I think, maybe will say more words on that.

CHAIRMAN POWERS: Thought you could get away sitting over there.

MR. PATTERSON: I'm Malcolm Patterson. I'm in the Office of New Reactors on the PRA staff, and I understand and sympathize with the concern you're expressing, but I think this particular applicant has dealt with it in a simple, perhaps onerous way, by providing their system level D-RAP list.

As I understand it, at this point all the reliability Assurance activities for all SSCs in those systems will be subject to D-RAP, until their panel removes a component.

MEMBER STETKAR: If that indeed, what you just explained, is everyone's current understanding; for example, if the COL were issued tomorrow, that indeed there would be in place the development of a reliability assurance program following the maintenance rule guidance for every SSC in every system in Table 17.4-2 and Table 17.4-3 of the COL FSAR.

If everybody is on board with that, I'd feel a lot more comfortable. If on the other hand at the time of the COL issuance, if it were issued tomorrow, the only table that I have of individual components is 17.4-1, if everyone believes that that is the master list --

MR. PATTERSON: That would be unacceptable.

MEMBER STETKAR: --for the reliability assurance program, I have real problems with that. So

I'd like to pretty clearly understand if the COL were issued tomorrow, what indeed is the scope of the reliability assurance program, from both the applicant's understanding and the staff's understanding, because you need to be able to perform a review and auditing against that scope somehow, if the COL were issued tomorrow.

MR. GIBSON: Can we respond?

MEMBER STETKAR: If you want to, yeah. I would hope --

CHAIRMAN POWERS: I mean it is my understanding, it was my understanding coming out that the list is a mechanism for identifying the systems, and it is the systems that are part of the scope now.

MR. HUGHES: Until such time as the systems are refined to the component level, the systems are in.

CHAIRMAN POWERS: So if it was issued today, every system --

MEMBER STETKAR: Every component in each, every one of those systems would be subject, would be on the RAP list.

MR. HUGHES: Are you threatening us with a license today?

(Laughter; simultaneous discussion.)

MEMBER STETKAR: As far as there's a liability to issue that license, yes.

MR. HUGHES: That's our intent, that's our understanding, and that's the plan. It can be revised prior to the license being granted, to refine it. If it were not, it would be at the stage it's at.

CHAIRMAN POWERS: Okay, and it was my understanding that this is just part of a systematic process, and it takes a while before you can do that refinement. The design has to progress a ways before you can do that refinement, and in that, you have three or four blocks of -- I can't remember how many up there, that said there's multiple refinements that take place before you get to the point that we say ahh yes, now we know what they're finally doing.

MR. GIBSON: Yeah, that's correct, and I'm sorry for our earlier presentations didn't make that clear to you.

CHAIRMAN POWERS: It made it clear to me.

MEMBER STETKAR: Okay. I just wanted to make sure that -- it was clear to me from the formal presentation. I just wanted to make sure it's clear to --

CHAIRMAN POWERS: Well, you've clarified half of it. The other half they're going to have to clarify. Yes. I mean we have a clear general understanding.

MR. GIBSON: Yes.

CHAIRMAN POWERS: Everybody seems to be clear on it.

MEMBER STETKAR: I'm good.

CHAIRMAN POWERS: I know that. This is an untenable situation. Somebody's got to be confused. Thank you.

MR. PHAN: With that, I'm going to Section 17.6, Maintenance Rule. Next slide please. At the end of Phase 2, there are two open items. The first one on Question 17.6-2, that we have the inconsistencies between the COL FSAR and the U.S. EPR FSAR section numbers.

The second ones are Question 17.6.3, regarding the implementation and readiness milestones for the CCNPP Unit 3 maintenance rule programs. Recently, the staff noted that the milestones provided in the Appendix A proposed combined license conditions of ten ITAACs of the COL submittals.

These questions have no impact on the technical contents of the maintenance rule. So they are conceded to be a minor issues, even though they are open items. That's our conclusion on 17.4 and

17.6.

MR. ROY: Any questions?

CHAIRMAN POWERS: Any remaining? Any questions on the overall?

MEMBER SHACK: Just thinking about this in the larger sense, I mean what you're really asking them to do is kind of a loaded 50.69, where they get to put in all the safety systems. Then they get to add all the risk-significant systems, but they don't get to take out any systems.

Could they actually do 50.69? Is that an option for --

MR. HUGHES: May I respond? I believe the regulations make it abundantly clear that any applicant under Part 52 can adopt 69 at their discretion.

MEMBER SHACK: Okay. So you could opt for that route?

MR. HUGHES: Yes indeed.

CHAIRMAN POWERS: Any other questions? Well, thank you very much. Thoroughly enjoyed the discussion, and yes, you outdid South Texas by a mile --

MEMBER SHACK: South Texas only did this for one system. They're going to do it for all the systems.

CHAIRMAN POWERS: yeah, but I mean I thought their presentation was edifying. I thought the staff presentation was edifying. I appreciate it very much. We're going to take a break until three o'clock, I guess, and then we're going to move to the --

(Off the record comments.)

CHAIRMAN POWERS: That's my understanding, okay. And we'll go into the rule of radiation protection requirements.

(Whereupon, a short recess was taken.)

Chapter 12, Radiation Protection

CHAIRMAN POWERS: We are about to launch into Chapter 12, Radiation Protection, and so Greg, I'm going to give the floor to you and I'll remind you that Mr. Ryan is a well-noted expert in the area of radiation protection. I might have a question or two.

MR. GIBSON: That would be excellent. Great, thank you. Our last chapter today is Chapter 12 on Radiation Protection. I'm joined with Tim Kirkham, and we are here to provide our overview of the reference COLA for CC3. In my introduction on page three, because again everything else that I said about incorporated by reference and so forth still applies.

For Chapter 12, we have taken no departures from the EPR FSAR. We have no ASLB contentions. We do have six COL information items that we'll be addressing and that Tim will be talking to.

We have four NRC SER open items and we'll provide the status on those, and then there are five NRC SER confirmatory items, and we'll be discussing those as well. So with that, I'm going to introduce Tim and if you could give your bio and vitae, I would appreciate that.

MR. KIRKHAM: Yes. My name is Tim Kirkham. I'm a senior health physicist, acting senior health physicist for UniStar. A little history about myself. This is my 29th year in health physics. I'm a Purdue University man. I started out life at Plant Hatch with Southern Company. Bounced around the Southern Company plants, then went to DOE Savannah River for five years.

Then Quad Cities Station as technical health physics manager, and then my last foray in the power reactor world was at Calvert Cliffs as technical health physics manager. Since then, I've been doing consulting.

Okay. My agenda today will be to discuss the COL items in Chapter 12, as well as any SER open and confirmatory items. COL Item 12.1-1 requests the applicant to describe our ALARA program, and to ensure that it follows the guidance in all the reg guides that are listed here, as well as Part 20 in the NUREG.

Our response is that the NEI 07-08 Alpha, which was approved by the NRC in October of 2009, and NEI 07-03 Alpha, approved in May 2009, is incorporated by reference into the Calvert Cliffs Unit 3 FSAR. Reg Guides 8.8 and 8.10 are addressed in 07-08 Alpha, and all the other listed guidance were used in drafting 07-03 Alpha. Any questions on this COL item? Fairly straightforward.

COL Item 12.2-1 asks the applicant to provide information on the site-specific sources that are going to be used, with activities greater than 100 millicuries. The next page shows the chart that's currently in the FSAR. I'm sure Mike's well aware of all these sources, but I'll go through them real quick.

The californium and antimony beryllium sources are used as start-up sources, and they are indeed clad in stainless steel for protection. The antimony beryllium source is not initially active, but becomes active once it's placed in the core, and then becomes activated during operations.

Two cesium sources are common with every reactor used, and these are used for the calibration of radiation protection instrumentation. These are commonly called Shepherd calibrators.

We do have a source in here that some folks like to ask questions about. We added the americium source, even though it's lower than 100 millicuries. We decided to list it, since it is a common source used for Alpha and low energy instrument calibrations. These americium are nickel electroplated.

The AmBe source is a neutron source used for calibration of neutron meters. Any questions about these sources?

COL Item 12.3-1 asks the applicant to provide information of how the guidance in these reg guides are used in the plant. These reg guides cover the gamut of air monitoring. 1.21 is measuring RAM and effluents and waste. 1.97 is accident monitoring. 8.8 is ALARA and ANSI 13.1 basically talks about sampling standards.

This item's addressed in 12.3 of the FSAR. All of the reg guides listed will be addressed via procedures and programmatic methods. Therefore, we have chosen to address this issue by stating our procedures and training will adhere to the regulations, the ANSI standards and the NEI templates.

COL Item 12.3-2 expresses interest in dose to construction workers. The next seven slides will address this issue. A slide layout is listed in the FSAR, that shows an overview of where the new reactor is with regard to the two current reactors, the independent spent fuel storage installation and the resin storage area. We'll show the layout on the next slide.

Three sources of exposure to construction workers are present at this site. Indirect exposure to gaseous effluents and then two direct exposures, one from the spent fuel storage installation and the other is the resin storage area.

The next slide shows an artist's rendition. It's a very good rendition, because it's hard to tell what's real and what's not.

(Laughter.)

CHAIRMAN POWERS: I'm not trusting the artists anymore.

MR. KIRKHAM: They're really good. If I might, I'll point out a few things here. Here's the current two units. Right here is where the new unit is going to go. Some of the things that we're going to discuss later on, sources of exposure, are right in here is the independent spent fuel storage facility, ISFSI is the way we refer to it. Right here is where the current resin storage area is.

We'll also point out a little later on when the -- as the new reactor is being built, this area down here is going to be used for parking, and this is going to be where our highest potential for dose is going to be coming across. That's right down here (pointing), because of these two sources. Any questions?

MEMBER RAY: Just a dumb guy question. Is that the best choice for a parking lot?

(Laughter.)

MR. KIRKHAM: Well, if I remember the parking lot, they're only going to be there for a few minutes a day.

MR. GIBSON: And also, there's a wetlands environmental issue. We want to minimize the impact on the environment, and wetlands is extremely valuable to us.

MEMBER RYAN: I see. What are the major fence line dose at the controlled area? I mean the closest point where somebody can walk after they park their car? I mean do you have a fence line dose projection at this point?

MR. KIRKHAM: The fence line dose here?

MEMBER RYAN: Yes.

MR. KIRKHAM: Is less than .05 millirads.

MEMBER RYAN: Less than .05 MR per hour?

MR. KIRKHAM: That's correct. Yeah, that is the resin storage area here, and so yeah. There's a fence here.

MEMBER RYAN: And that's the resin storage area for the existing plants as well as the new plant; correct?

MR. KIRKHAM: That's correct.

MEMBER RYAN: So that will be an integrated exposure.

MR. KIRKHAM: That's correct.

MEMBER RYAN: How do you expect the volume to increase and the dose rates to increase over time?

MR. KIRKHAM: Right now, we don't expect it. We expect it to ebb and flow. There's contracts in place to get rid of the resin that's already there.

MEMBER RYAN: So you're accumulating just for the purpose of transport and disposal?

MR. KIRKHAM: That's correct.

MEMBER RYAN: Okay.

MEMBER STETKAR: You mentioned that's a construction parking lot. Where's the permanent parking lot for the site after it's finished?

MR. KIRKHAM: Well, I don't see that on the drawing up there.

MEMBER STETKAR: They've got to walk.

(Laughter.)

MR. KIRKHAM: That's a good question. As I remember, some of the permanent parking is out here, according to the -- is it on the next slide? Go to the next slide. Let's see, where are we here? Yes. There's going to be another picture after this, but right here is two blow-ups. This is of the current front unit.

Right here is where ISFSI is, and then right here is the resin storage area. So right there is the construction. Actually, I called that a parking area. It's actually a laydown area, for a construction laydown area. Here's the parking here. Laydown area. I don't see the parking area listed there either.

(Simultaneous discussion.)

MR. KIRKHAM: Construction parking here and here, further away from the current dose, from the

ISFSI and the resin storage area.

You can also see that the new build here is about 1,000 feet from the shore. Up here is the shore. So therefore, when we did the calculations for the construction workers, we did not take into account any doses from liquid effluents, because it usually can't get there. I think that's all I wanted to point out on this picture.

Next slide please. This is a blow-up of the existing plant, and again, the new units are going in over here, and the spent fuel storage facility and the resin storage facility are down here. That gives you an idea where we are.

Next slide. Here we are, the spent fuel storage facility and the resin storage facility here.

MEMBER RYAN: Am I lined up on the previous picture, if I look at that upper left switchyard, and that's the switchyard kind in the lower middle of the previous picture?

MR. KIRKHAM: Yes, that's correct.

MEMBER RYAN: Okay, great.

MR. KIRKHAM: So I hope that helps.

MEMBER RYAN: It does.

MR. KIRKHAM: Get the feeling of where things are on site. The other thing that's shown in here is the original steam generator storage facility. There are currently used steam generator storage there. But again, the dose rate on the outside of that is less than .5 millirad. That's under surveillance by the current Unit 1 and Unit 2.

MEMBER RYAN: There's one steam generator there or --

MR. KIRKHAM: Two.

MEMBER RYAN: Two. What's the plan there? Are they going to sit a while?

MR. KIRKHAM: Yeah. The last thing I heard, they're going to wait until decommissioning. That's the last thing when I was -- when we actually moved them. Actually, I guess there were four steam generators in there. I'm only thinking of one unit. There are four steam generators in there.

MEMBER RYAN: Four steam generators, and you're going to hold onto them for another 20 years?

MR. KIRKHAM: Yeah. I don't know what the current plant management thinking is on liability of keeping steam generators. Again, when we first, when we took the first two out, the idea was we're going to wait and see what technology does. You know, maybe it will decontaminate them, maybe it will melt them down. It would be nice to be able --

CHAIRMAN POWERS: Sell them to Argonne for test materials.

MR. KIRKHAM: There you go.

(Simultaneous discussion.)

CHAIRMAN POWERS: Just kind of help Shack out.

MR. KIRKHAM: Any other questions on the general layout? Next slide, where we start talking about --

MEMBER RYAN: I'm sorry. Just one last question. Where's your nearest resident?

MR. KIRKHAM: Nearest resident?

MEMBER RYAN: Resident, I mean member of the public.

MR. KIRKHAM: Let's go back to --

MEMBER RYAN: The aerial photograph maybe, 12?

MR. KIRKHAM: Go back to Slide 12, yeah. The nearest resident is, let me get my bearings. Out over here.

MEMBER RYAN: Somewhere off of that circular road that goes around?

MR. KIRKHAM: Yeah.

MEMBER RYAN: So give me a hint. Two miles, a mile?

MR. KIRKHAM: Since I haven't been there, it's more than a mile.

MEMBER RYAN: Somewhere between one and two maybe.

MR. KIRKHAM: Yes.

MEMBER RYAN: Just guessing.

MR. KIRKHAM: Yeah. It would be more than a mile.

MEMBER RYAN: And the nearest population center of any size is?

MR. KIRKHAM: Is Lusby, probably 15 miles south.

MEMBER RYAN: Okay, thanks.

MR. KIRKHAM: Okay. Again, the COL asks us to list basis and models and assumptions and results, so we're going to go through that real quick here, so you can see that's normally the part that everybody cares about, is dose to construction workers. So we'll spend a little bit of time on that.

The basis is that there are three main sources of exposure, like I mentioned before: gaseous effluents from Units 1 and 2, and then two sources of direct exposure. The models that were used in the calculation use undepleted, undecayed gaseous effluents. Those were used for calculating the gaseous.

Then the direct dose was calculated also based upon a 100 by 100 foot squares. The whole site was divvied up into a matrix, and then assigned doses from each source were assigned to each of those 100 foot squares, and then summed up.

Next slide. The assumptions were that the effluents do not change. Values from 2006 were used, because that's the highest in recent data. We also assume that the ISFSI campaigns stay the same, same number of fuel shipments from the plant up to the ISFSI storage.

MEMBER RYAN: So that's kind of a linear increase over time?

MR. KIRKHAM: Yeah, up to --

MEMBER RYAN: Including decay, of course.

MR. KIRKHAM: Right, right, right. The calculations were done out to 2015, assuming that that's around about when the time frame we're concerned about, and we also assumed that the resin storage remains stable. Like I said before, it's going to ebb and flow. There's going to be some resin stored up there, and then they'll ship it off to Studzik (ph) or somebody to go do BR.

Inputs were REMP report data, REMP being

Radiological Environmental Monitoring Report, and the data from annual effluents reports. Those were the data that's used to calculate doses.

Next slide, Results. Tables were made, and the tables are in the application, that show the occupancy factors for various locations, i.e., based on those 100 by 100 foot sections. Then tables also show the dose rates at various locations.

Projected annual dose is less than three millirem per year for a 2,200 worker year. The highest dose that someone could get is 39 millirem per year, and that's the proverbial fence-setter.

MEMBER RYAN: So that's the highest, and the average is three you said?

MR. KIRKHAM: Average is three, and that takes into account the occupancy factor in different zones that were used.

MEMBER RYAN: Okay.

MR. KIRKHAM: And all that compares to the average annual background dose is 52 millirem a year. So we'll be adding, what, five percent?

MEMBER RYAN: 52 is just external though?

MR. KIRKHAM: Yes. And then we do have an ALARA agreement. It's outside of the scope of the FSAR, but there is an ALARA agreement that states that

Unit 1 and Unit 2 is still placing surveillance TLDs with personal TLDs and REMP TLDs around the area, around the construction area, and those are going to be analyzed on an annual basis, to make sure that all of our calculations were indeed correct, and that we're not going to over-expose any construction worker.

MEMBER RYAN: You think annual is enough?

MR. KIRKHAM: Well, actually they're collective more than an annual, more than on an annual basis. So they'll be getting data on about a six month basis.

MEMBER RYAN: So you get data on a six month basis because you're doing an annual report?

MR. KIRKHAM: I can't say an annual report, but there will be an annual review one. Probably a report's going to be written, but that wasn't --

MEMBER RYAN: I guess it's not much of a review without a report.

MR. KIRKHAM: Yeah. Well, you've got to be careful how you define report, who reviews it --

MEMBER RYAN: Assessment document. How's that?

MR. KIRKHAM: An assessment document, good point.

CHAIRMAN POWERS: Do the existing plants, when they anticipate evolutions, will they do their ALARA analysis, recognizing that there's a construction activity taking place?

MR. KIRKHAM: Yes, they will, and largely the only evolutions that are going to occur that's going to affect the construction workers will be moving a canister of fuel, because they'll go up a road that takes them around towards the construction, and they'll use the same road for moving waste, resin or filters.

MEMBER RYAN: Do you have any big outages that might affect the construction activities that are planned or on the horizon?

MR. KIRKHAM: I think all the big stuff's been done. All the steam generators have been replaced. I don't know what else is left that would create a -- plus the other good thing is whenever we're moving a high dose rate item up that road, there's always a RP tech that accompanies them, to make sure that people are run out of the way.

MEMBER RYAN: So it sounds like any interruption to schedule might be temporary and minor? Is that your assessment?

MR. KIRKHAM: Yeah. It would be on the terms of minutes as opposed to days, yes.

MEMBER RYAN: Okay.

MR. KIRKHAM: Any other questions on construction worker dose?

(No response.)

MR. KIRKHAM: Okay. Slide 19. This COL item is also addressed in 12.3.4.5. The applicant was asked to describe instruments and training for determining iodine concentrations during an accident.

First, NEI 07-03 Alpha describes this issue and is already incorporated by reference. however, we have chosen to address it further by stating that a portable monitoring system will be used that meets NUREG 07-37 requirements, which says that we'll have the capability to move the cartridge to a low background area, low contamination area, and that we'll also have the capability to monitor iodine present during accident conditions, and that we'll have sufficient samplers to sample all vital areas.

Health physics personnel will be trained on the use of these monitors and how to interpret the data via the RP training program. The in-plant system will be able to determine iodine concentrations in the areas most likely to have this issue, containment, perhaps penetration rooms, all of that.

Next COL item asks us to describe the RP program, and says that it must comply with all the applicable reg guides and CFRs. Our response is shown in the next slide. We've addressed this COL item by incorporating by reference of the approved 07-03 Alpha.

In that document, there's two bolded items in that document that we have to provide site-specific information for. The first one talks about access of the very high radiation areas. There are very few -- it will ask us how we're going to control and ask us to identify the VHRAs.

So identification is during normal operations, these very high radiation areas are the spent fuel storage pool, which is flooded; the spent fuel storage pit, which is also flooded all the time; the transfer pit, which is flooded, and then there's a fuel transfer tube area that is only a VHRA during fuel moves. So very few areas we have to worry about.

As far as control is concerned, they're going to be typical power plant controls. So there's going to be positive control access keys that are controlled by one individual on site, the radiation protection manager, and then an associated sign-out log, so we know who's got the key, how long they've had it, where they are. That's the control.

CHAIRMAN POWERS: Any of those correspond to fire areas?

MR. KIRKHAM: No.

CHAIRMAN POWERS: No combustibles?

MR. KIRKHAM: No, that's correct. Good question though. Then the second part is we were asked to address a quality assurance program, and that's all addressed in Section 17.5, and the quality assurance program was approved by the NRC in 2007. That was discussed earlier. That's where the RP QA program resides.

Next we'll briefly cover open -- very briefly cover open and confirmatory items.

MEMBER RYAN: Tell me when you're going to get to 20.1406. Is that coming?

MR. KIRKHAM: That is not coming. Well, that's covered in some of the open items.

MEMBER RYAN: Okay. I guess we'll get to it then.

MR. KIRKHAM: Yeah.

CHAIRMAN POWERS: That's tough to beat.

(Laughter.)

MR. KIRKHAM: This slide shows our current

SER open items. The response to all these open items have been submitted to the NRC for review, and will be included in the next revision of the FSAR.

MEMBER RYAN: See, you did get there. Number four.

MR. KIRKHAM: Buried pipe?

MEMBER RYAN: Yeah.

MR. KIRKHAM: Yes. Ask away.

MEMBER RYAN: Well, I mean, you know, the news is certainly out there today for a couple of plants that are addressing issues that are related to what 1406 is trying to address. So you've got a situation where you've got an existing plant, and two existing plants and more units to come.

How are you going to, you know, look at the system, the whole site? Because 1406 isn't just by reactor new and old; it's by site. You know, the reactor part of the license for the machine is one aspect, and then of course, you know, marrying it to a site and a site-specific application is the second part.

So even though it's a little early for you to be thinking about it, you do have an existing site and you should be thinking about it.

MR. KIRKHAM: It's not early to be

thinking about it. You're right, Mike. The current units have already been through the 1406 and looking for pipe, and back a few years ago they did have an issue of a sink hole and a pipe that they didn't know had been leaking.

The only thing that they found there was tritium. That was the only thing left. But that's been -- that sink hole's been fixed. The pipe has been taken care of. We're also, we have the advantage for both Unit 1, Unit 2 and Unit 3, is that there's a very thick layer of clay that was placed underneath the reactors, in the bed level of it.

So if there's any water that does get a leaking pipe or anything, it won't go into the ground water due to that thick level of clay.

MEMBER RYAN: How about if it runs off that clay?

MR. KIRKHAM: If it runs off that clay --

MEMBER RYAN: Which is what it will do. It will run off the clay and then into whatever's adjacent to it.

MR. KIRKHAM: Units 1 and 2 currently have several monitoring wells that looks for that. Unit 3 will also have monitoring wells looking for that. Also, towards that end, the buried piping for Unit 3 is all double-walled concentric pipe in a pipe.

MEMBER RYAN: And is it instrumented for leaks?

MR. KIRKHAM: It's instrumented for leaks and --

MEMBER RYAN: I'm sorry. You said that will be for unit, for the new unit?

MR. KIRKHAM: For the new unit.

MEMBER RYAN: But not -- there's nothing like that in the old units, I'm guessing?

MR. KIRKHAM: I don't think that's been retrofitted with a pipe in a pipe.

MEMBER RYAN: Okay. Some places they've addressed that, and some not.

MR. KIRKHAM: Yeah. I don't know for sure whether they have. It's been a while since I've looked at that further. Any other questions on that?

MEMBER RYAN: Not so far.

MR. KIRKHAM: Okay. Next slide. NRC SER confirmatory items are listed here. I'm not going to go through them, but the confirmatory items will be reviewed by the NRC, and these minor items have already been approved by the NRC, and we just need to incorporate them into the FSAR. That's my part of the presentation.

MEMBER STETKAR: Tim, let me ask you, since you brought up buried piping and we're doing okay on time, at the DCD presentation on I think it was Chapter 12, I asked a question about -- I know that the pipe in a pipe applies for all liquid effluent release lines.

I asked a question about any buried pipe that may connect the reactor building or waste handling building together. In other words, intra-built, inter-building buried piping, whether there was any of that at the site, and whether that also had the same type of piping construction.

MR. KIRKHAM: Pedro, can you answer that.

MEMBER STETKAR: Pedro took it away, and I saw Pedro sitting there. So I thought maybe he came back with an answer.

MR. PEREZ: Okay, good afternoon. My name is Pedro Perez with AREVA, and you're correct. I did take away about five or six questions, and one of them was about what happens with these interfacing systems. We do have a requirement that we think the interfacing systems, if there's something that needs to be at the lowest elevation now, at the very lowest elevation below the floor.

Obviously the drain, you know, the drain, those will have to be double-wall with leak protection. Same thing with sumps.

MEMBER STETKAR: Okay.

MR. PEREZ: At the lowest point.

MEMBER STETKAR: Yeah, yeah. So anything below bottom basement floor level.

MR. PEREZ: Yes, that would have a direct path into the environment.

MEMBER STETKAR: Right. Okay, great. Thank you.

MR. GIBSON: Okay. Well with that, again we have no departures from this particular section. We've had no ASLB contentions. We've got four NRC open items, and we have provided responses to those. The five confirmatory items will be drilled into -- okay.

Most of the -- all five of the confirmatory items will be incorporated into Rev. 7, which is coming out in September. So we will have that. I just got a note of clarification with regard to the parking. Most of the construction workers are going to be parking off site and being bused in is the current plant.

MEMBER RYAN: Oh, there you go. That's an update. Thank you.

MEMBER STETKAR: So that area, you said, is adjacent to the resin storage facility is a laydown area.

MR. GIBSON: Is a laydown area, that's correct.

MEMBER STETKAR: But would be accessible.

MR. GIBSON: Yeah.

MEMBER STETKAR: Yeah. But there might be people in that are more frequently than just a parking area, if it's a fabrication area. You've taken care of that?

MR. KIRKHAM: I think according to the occupancy factor, they figured two percent of a worker's time could be in that area, which is probably conservative.

CHAIRMAN POWERS: A material handler might be there that much time.

MR. PORCHET: May I? I'm Rob Porchet. I'm the Reg Affairs Project Manager for Calvert 3.

CHAIRMAN POWERS: You need to introduce yourself again.

MR. PORCHET: I'm Rob Porchet. I'm the Reg Affairs Project Manager for Calvert Cliffs Unit 3, and a question was brought up earlier about the nearest resident. The nearest resident is 3,000 feet, and there are 30 resident within one mile.

MEMBER RYAN: Oh are there?

MR. PORCHET: Yes.

MEMBER RYAN: Okay. Thank you.

CHAIRMAN POWERS: Any other questions on this? Surinder, we're ready for your troupe.

MR. ARORA: We are.

CHAIRMAN POWERS: Thank you very much.

MR. KIRKHAM: Thank you.

CHAIRMAN POWERS: And my compliments to your artist.

(Laughter.)

CHAIRMAN POWERS: But I'm never going to trust another picture you show me.

(Pause.)

CHAIRMAN POWERS: The committee is being treated to a special treat here, because Sara is a graduate of an outstanding course on Perspectives in Reactor Safety. So I know that she will give an extremely educated and refined presentation before the committee here.

MR. WIDMAYER: Did she get an A?

CHAIRMAN POWERS: Yeah, you bet you she got an A.

MR. WIDMAYER: Those don't come easy.

(Laughter.)

MR. ARORA: Again, there is Jason Jennings as the chapter PM for Chapter 12. Unfortunately, Jason had to leave today, and we have another Jason substituting for Jennings. He'll take care of the presentation, Jason Carneal.

MR. CARNEAL: All right. Good afternoon. I'm back from the presentation this morning on Chapter 4. I'm Jason Carneal, and I will be taking Jason Jennings' place for this presentation.

During this presentation, we'll give the staff safety evaluation to date on Chapter 12 of the Calvert Cliffs Unit 3 combined license application. Technical staff involved in this review are Sara Bernal from the Health Physics Branch.

In Chapter 12, we've issued a total 12 questions, and four of those questions remain open items. All of the open items that are specific to Chapter 12 are in sections 12.3, 12.4, Radiation Protection Design Features. There is site-specific information in the entirety of Chapter 12.

This slide gives an overall description of the open items that we've identified in our safety review. They're all contained in RAI 199 and again, they affect Section 12.03 and 12.04. They deal with the ALARA program, conversion errors between rem and sieverts, which may be of interest to the committee, buried piping.

MEMBER RYAN: Especially since it's going to be a factor of 100 the wrong way.

CHAIRMAN POWERS: It's something I do every time I make the train conversion.

MR. CARNEAL: And design features and monitoring for vacuum breakers. For the details of the open items in Chapter 12, I'll turn the presentation over to Sara Bernal.

MS. BERNAL: Okay. Good afternoon. My name is Sara Bernal. I was the lead technical reviewer for Chapter 12, and I was supported in my review by the Health Physics Branch. Section 12.1 --

CHAIRMAN POWERS: Oh Sara, we've got a rule here. You can't start your presentation without telling us something about yourself.

MS. BERNAL: Well, I heard you only have to do that once.

(Laughter.)

CHAIRMAN POWERS: We need to be reminded.

MS. BERNAL: Okay. Well, I attended Dr. Powers' class.

(Laughter.)

CHAIRMAN POWERS: And enjoyed every minute of it, right?

MS. BERNAL: Okay. For Section 12.1, I'll just get into it. I have a Mechanical Engineering degree from the University of Michigan. I have Master's in Nuclear Engineering from the University of Michigan. I've been at the NRC for five years as part of the Health Physics Review Branch.

I'm a qualified technical reviewer and that's pretty much resume. Again, I was supported by my technical branch in this review, and that's it for me.

Section 12.1, again the staff reviewed the applicant's description of the ALARA program. The applicant's FSAR endorses NEI 07-08, which is a generic description of the ALARA program. The applicant also endorses NEI 07-03, which is a generic description of the radiation protection program, which also describes ALARA program components.

There are SERs written on both templates. Therefore, the adoption of these templates into the applicant's application is acceptable to the staff. The applicant's endorsement of the NRC-approved version of these templates is being tracked as a confirmatory item until the FSAR is revised.

In Section 12.2, the staff reviewed the applicant's response to one COL information item. The applicant was required to provide a description of site-specific sources that were above 100 millicuries. The applicant provided a table, including all start-up and calibration sources, some of which did not meet the 100 millicurie threshold. So they provided a very complete table.

RAIs were asked regarding the lack of a neutron instrumentation calibration source, as well as RAIs were also asked regarding the security requirements of 20.1801 for sources. In addition, an RAI was asked on the compliance with the National Source Tracking requirement of 20.2207.

The applicant's response to these RAIs included a reference to EPR Figure 12.3-16, which is a figure of elevation 0 of the auxiliary building, which includes a source storage room. The staff determined that this room and this drawing is sufficient to support a conclusion of control of the sources and compliance with 20.1801.

In addition, the applicant revised their FSAR to include a neutron instrumentation calibration source, the AmBe source, and finally the applicant stated that they would comply with the source tracking reporting requirements of 20.2207 for the 400 cesium or 400 curie cesium 137 calibration source, which is the only source that triggers this requirement.

MEMBER RYAN: And that's the one we heard about earlier, that's under key control by one person; correct?

MS. BERNAL: Well, that was about very high radiation areas.

MEMBER RYAN: That was access to one source, I thought. Maybe I misunderstood.

CHAIRMAN POWERS: I understood it to be high radiation areas.

MR. ROACH: This is Ed Roach. I'm the branch chief for Health Physics, New Reactors. I believe what Mr. Kirkham was talking about was the controls, the programmatic controls for high radiation.

MEMBER RYAN: Oh, very high radiation. You're right.

MR. ROACH: If sources actually create a high radiation area or a very high radiation area, then those same controls would apply to those in the course of --

MEMBER RYAN: Tell me a little bit about the access to high source, a large source like this?

I mean how is that controlled?

MS. BERNAL: Usually, there is -- well, the applicant, in terms of this application, they commit to the guidance of Reg Guide 8.38, which describes access, adequate ways of conforming with 24.1601, 24.1602, so that they will control access to the switches, in compliance with the regulations.

MEMBER RYAN: Okay.

MS. BERNAL: Okay. Next slide please. Section 12.3 and 12.4 has three COL information items. The first information item states that the applicant should describe the application of these regulatory guides, and the application of ANSI 13.1 to sampling, recording and reporting of airborne releases of radioactivity.

As the applicant stated earlier, they will use the above guidance to develop procedural criteria and methods for obtaining representative measurements of radiological conditions, including airborne radioactivity.

Radiation protection personnel responsibilities as far as radiation surveys, as well as types and frequencies of surveys were also described in the FSAR of the application. These descriptions of survey types and frequencies and responsibilities conform to the guidance of Reg Guide 8.2 and 8.10 and support compliance with 10. C.F.R. 20.1501, and therefore is acceptable to the staff. There are no open items related to this COL information item.

Next slide. The second COL information item is the dose to construction workers. Again, as the applicant stated, dose to construction workers were calculated based on liquid and gaseous effluents, and the direct radiation sources on site.

Staff issued RAIs requesting the applicant to describe how doses to construction worker would be monitored to ensure compliance with ALARA, and also the public dose limits of 20.1302. The applicant replied that the radiation protection program and ALARA program for the existing units would be extended to include these construction workers.

The programs of Units 1 and 2 currently comply with requirements of Part 20, and therefore the staff considered this to be acceptable. As stated earlier, the staff also issued RAIs on some unit conversion errors within the FSAR. The applicant has corrected these errors and this open item is now confirmatory.

The third COL information item in this

section is the compliance with the requirements of 10 C.F.R. 5034(f)(2)(xxvii), which is -- and also conformance with the criteria of NUREG 07-37, which is post-accident iodine monitoring.

The applicant stated that their instrumentation would conform to the guidance in the NUREG. The staff finds this to be acceptable. The applicant also stated that emergency procedures would be developed for the measurement of post-accident iodine concentrations, and that the radiation protection program would ensure appropriate training, maintenance of equipment, as well as procedure implementation. This conforms with SRP and NUREG 07-37, and is therefore acceptable.

Next slide. In addition to the COL items, we also asked the applicant to describe their compliance with site-specific requirements, COLA requirements of 10 C.F.R. 20.1406. 20.1406, of course, has an operational component.

The applicant was asked to describe site-specific design features that would demonstrate compliance, and specifically was also asked to address any buried piping that could potentially become contaminated, as well as to call out and address any vacuum breakers associated with the effluent discharge line that could exist, which is dependent on the site.

MEMBER RYAN: Does piping include electrical piping, piping that contains electrical conduit? If it penetrates the plant, that's an access pathway, as well as water pipe.

MS. BERNAL: I did not ask the question when I asked for contaminated, piping that contains contaminated liquids.

MEMBER RYAN: But it's interesting to think about, you know, the system of ground water contact with the plant itself in a number of different ways. It's going to cycle through the unsaturated zone. The saturated zone is all connected, and if there's an open conduit carrying underground electrical cable, why couldn't that be a conduit too?

Just it's something to think about. I've been thinking and studying a bit about 1406, and I'm not convinced we've got our hands around all the pathways yet.

MS. BERNAL: Okay.

MEMBER RYAN: It's interesting that a couple of plants have had that pop up unexpectedly in the last six months or so.

MR. ROACH: Dr. Ryan, this is Ed Roach, Chief of the Health Physics Branch, and related to

20.1406, that basically is incorporated by Part 52 for the new plants being built, of which this plant is one of those.

MEMBER RYAN: Right.

MR. ROACH: The guidance that was presented for that included Reg Guide 4.21, that gave examples of penetrations within the building structure, that could allow excess or egress, including building seams in conduit or piping. I believe in the example --

MEMBER RYAN: Well, it is on the table.

MR. ROACH: It's back in the category for the examples that are in there. Additionally, we've been monitoring the operating experience of all the current generation fleet. The current generation fleet is adhering a guideline, a voluntary guideline known as NEI 07-07, which had voluntary compliance.

When we implemented the 20.1406 with new reactors, NEI came up with a newer, more in-depth document, NEI 08-08(a), which was endorsed in a safety evaluation report. That carries with it the requirements that it basically becomes an operational program, where they assess the risk to the site for that facility, looking for the most likely failure points, and then mitigating the potential --.

I think there's a step change in the quality. But we continue the operating experience because of that.

MEMBER RYAN: And I guess the question for today's discussions, is that kind of thinking being integrated for the new unit here?

MR. ROACH: Yeah. I would say from our approach, we think they've been responsive. I know there's still an RAI on the table --

MEMBER RYAN: Oh, so that's a work in progress maybe we'll hear a little bit more about later on.

MR. ROACH: Yeah.

MEMBER RYAN: Okay, thanks.

MS. BERNAL: There's an RAI out to the design certification also on buried piping. Okay.

MEMBER RYAN: I think, and just a follow-up comment. I mean I think it's important that other kinds of penetrations are addressed, and not just those that carry liquid. But any pipe that's joined somewhere inside is a potential leak point or ingress point of water coming in and washing back out and cycling in and out, who knows what, based on the geohydrology and the aquatics of the site.

You know, it's not just the radioactive

material. It's the water. What's the water doing? So that's something to think about. But I'm glad to hear that's being integrated.

MR. ROACH: This is Ed Roach again. There are operating experience examples of where the facility sits in an area that has the water level, ground water level fluctuates, and such it actually gets, occasionally gets in-leakage into their auxiliary building, and then subsequently out-leakage.

MEMBER RYAN: You know, the other examples that we can think about for 1406 are decommissioning examples, of how those lessons learned are incorporated. I guess that's sort of the guidance really.

MR. ROACH: Actually, in Reg Guide 4.21, the lessons learned from Maine Yankee, and they were incorporated and collected and then summarized and put into that table.

MEMBER RYAN: Okay.

MS. BERNAL: Thanks, Ed. No more questions on that. Okay. Finally, in Section 4.3, the applicant was asked to address some conceptual design features that are in the EPR FSAR, with regard to the access building. The access building is a conceptual design in the design certification.

But it includes several facilities which are important to the radiation protection program. The applicant was asked to address this conceptual design information, because it is out of scope of the design certs. So the COLA applicant needs to take care of it.

In response to the staff RAI, the applicant incorporated the EPR design conceptual information with regards to these radiation protection facilities. The level of detail that they incorporated is acceptable and conforms to the guidance of Reg Guide 8.8 with respect to support facilities for the radiation protection program, and also NEI 07-03 describes these support facilities.

So compliance for this access building description within their application, combined with the incorporation of NEI 07-03, satisfies the staff requirements with regards to a description of radiation protection program support facilities.

Next slide. Section 12.5 again is a description of the operational radiation protection program. The staff endorsed, or I'm sorry, the applicant endorsed NEI 07-03. There was an SER written on this template. Therefore, its adoption by the applicant is acceptable to the staff.

The applicant's endorsement of the NRC-approved version of the template is being tracked as a confirmatory item. This will become closed once the application is revised. NEI 07-03 contains two bracketed sections which request site-specific detail.

The applicant provided descriptions on very high radiation areas, including access controls and also address the reference to the quality assurance program that would cover the radiation protection program. These descriptions conform to the guidance of the SRP, and therefore are acceptable to the staff for demonstrating compliance with 20.1101 as it relates to periodic auditing in the radiation protection program.

The access control requirements conform to the guidance of Reg Guide 8.38, and therefore satisfy the regulations 20.1601 and 20.1602. That's all, I guess.

Conclusions. This is the standard conclusion slide, open items and confirmatory items. Therefore, we're unable to finalize our conclusions. Is there any questions?

MEMBER RYAN: I have no questions.

CHAIRMAN POWERS: Any additional questions on this?

MEMBER RYAN: No.

CHAIRMAN POWERS: I think we're done with this presentation then. Thank you. Are you in a position to hear from Mr. Cook?

MR. WIDMAYER: I think we are.

CHAIRMAN POWERS: We have a presentation from Mr. August Cook?

MR. WIDMAYER: No, Jim August.

CHAIRMAN POWERS: Jim August, I'm sorry.

(Off the record comments.)

CHAIRMAN POWERS: Mr. August, whenever you're ready. Let's see if you're going to stand, you need a mobile microphone, I think.

MR. AUGUST: Where is your microphone? Where is your microphone?

CHAIRMAN POWERS: Those black ones sitting on the bench. So if you sit down, you're fine.

MR. AUGUST: I'll sit down. I'll sit down.

MEMBER SHACK: If you speak really loud, it's okay.

Presentation on Reliability Assurance Programs

MR. AUGUST: Well, I'll lead in by telling you I'm slightly concerned I might lose my voice, because I had a cold two weeks ago, and as you know, you get a dry throat after that. So I'm not contagious at this point in time, my wife tells me, but I will caution you. I'm a little bit worried, which is why I have this here.

With that caution, my name is Jim August. I heard the chairman mention earlier he'd like to hear a little bit about folks in advance. My background is I've been in the nuclear industry since about 1982 commercially. Prior to that, I've got about six or eight years with the U.S. Navy.

My involvement in reliability really came about, though, as a result of commercial activity beginning in 1982, hired to improve reliability on some highly unreliable equipment at a nuclear power plant, which some of you folks would recognize, called Fort St. Vrain, which was ultimately commercially shut down because of unreliability issues.

So since that time, I spent the balance of my career doing reliability type work in a variety of different contexts, but primarily nuclear context. My background is I've got a degree in Physics, a degree in Mechanical Engineering and another, a Master's degree in Mechanical or in Engineering.

I'm a professional engineer in the state of Colorado, and like I say, I've done a lot of this work for many, many years. So with all that, this presentation is titled "Why Certify New Nuclear Plant Maintenance Programs?"

It stems from a white paper that I believe has been forwarded to the ACRS. It's titled "Nuclear Maintenance Certification, License Plants With a Plan." It was written by me and reviewed by an associate, Joe Hunter, who's an SRO.

To summarize, what this paper talks about, which I'm going to present here, is first of all, why you have reliability assurance programs. Secondly, how it ties in with Part 52, what the intent of Part 52 is, what is an effective reliability assurance program, what the benefits of an effective reliability assurance program are, why now is the time to think about those, and what we should do.

To sort of summarize, after 40 years, nuclear plants need scheduled maintenance plans. Earlier, there was this handout that was provided by Calvert Cliffs or pardon me, by UniStar, associated with Calvert Cliffs 3 and 4. I couldn't have asked for a better sort of straightman-type presentation, because it's got this line here and it shows D-RAP and O-RAP, and sort of a gap in between.

Well, really what I'm talking about here is that area in between. So first of all, to come back, why reliability assurance programs. Well, Part 50 experience is why. Part 50 was characterized, or the plants, if you will, that were licensed under Part 50 were characterized by ineffective maintenance and unreliable equipment.

There were many legacy problems, delays and uncertainty, but certainly plant reliability was one of the biggest. It was important before Three Mile Island, after Three Mile Island, when you started seeing the reports like NUREG 646 and 737, the Kemeny Report, the Rogovin Report, it was very clear reliability may have been the most central issue in the entire nuclear industry at that time.

Even after Three Mile Island, you had a continuing release of generic communications from the NRC that lasted about 15 years. If you go back and look at these now, they're around 300 of these various generic communications, out of a grand total of 1,000 that address power reactors, that in some way, shape or manner address design basis reliability concerns.

So to summarize, the big problem with Part 50 was that deterministic design did not assure reliability, did not assure reliability at all. So we get to Part 52. Part 52 was the attempt to correct the weaknesses of Part 50. A very good way to summarize Part 52 is it took a very different approach to design.

Whereas Part 50 was a design, build, license approach, Part 52 is a design, license, build approach. Okay. So Part 52 tried to rectify and correct the weaknesses of Part 50, where you asked for the license at the end of the day, if you will.

Part 52 requires a reliability assurance program. Again, the big difference with Part 50 is Part 50 assumed that reliability would just sort of happen, and it didn't. So the reasons that Part 52 requires a reliability assurance program is to assure the reliability in the PRA. Reliability is certainly required for operations, but it's really the design basis for certification from the PRA.

In other words, if your PRA specifies certain levels of reliability and availability for equipment and you don't meet that, you're really out of the basic design certification for that plant. That was recognition, and that was why the provision was put in for reliability assurance programs.

Ideally, Part 52 would answer. It would say what provides an effective reliability assurance program? What meets the intent of a reliability assurance program effectively? What specifically fills the so-called design reliability assurance program or D-RAP, and the operational reliability assurance program or O-RAP?

Unfortunately, Part 52 doesn't really answer those questions. It's incomplete today. It's very reasonable to require a reliability assurance program, even putting aside nuclear issues, where it's clearly required for the PRA.

Designers should provide guidance for safety-related SSC. Designers have provided that for other applications. So they should just provide it. It's very well-established that the responsibility for reliability on a piece of equipment in a very general sense under the law, under English common law, rests with the designer. So this continues the intent of Part 52.

Meeting Part 52's intent would leave much less to chance. So the real question, I believe, is what is a reliability assurance program? Well, to answer that question, you have to sort of answer what is the intent of a reliability assurance program?

Well ideally, a reliability assurance program would assure nuclear systems, structures and components or SSC, operate with minimum unavailability and the fewest maintenance preventable function failures or MPFF, just as required by the maintenance rule 50.65.

Clearly, it should assure nuclear plant SSC meet design-assumed reliability and availability. It should provide, and this is the kicker, it should provide actionable guidance to those who operate and maintain the plants.

The difference between an effective program and what we presently require is what I call the RAP gap. You get that difference by simply doing a gap analysis. What do I need to get me here? What do I have today? What the RAP gap does is question the effectiveness and adequacy of the rules or the guidance, if you will, under Part 52 for a reliability assurance program.

What would be effective? Well, to be effective, a reliability assurance program would address the certified design, as well as the combined license. It would provide tasks that actually make SSC reliable. It would give good guidance. It would complete the design. That is to say it would not just be a list of equipment, which is what I've heard a lot of discussion so far earlier today.

It would be complete actionable guidance.

It would specify activities. It would define specific tasks that make SSC reliable. Those tasks and activities would have to be clear, explicit, actionable and measurable.

What this really is is simply equipment scheduled maintenance plans, completely specified according to standards, as part of the certified design. Now I know what you're thinking right now. You're thinking well how could you possibly do this?

Well, to do this, you would have to have an effective consensus-based SSC scheduled maintenance plan development process for complex safety equipment and designs. It would have to provide an effective scheduled maintenance program that in essence would become the reliability assurance program.

That would have to come complete with actionable guidance that would assure its performance. It would have to be something that had been proven over time, tested with qualified systems and participants.

Nuclear plant programs today could specify a much more effective reliability assurance program simply as a complete scheduled maintenance plan. In fact, there's a standard that tells you exactly how to do this. It's called MSG-3. The last version was issued in 2004. It was first issued about 1968 as MSG-1. It's called Scheduled Maintenance Development Process.

It's a documented, well-proven method for developing reliability programs and complex safety applications that was developed in the aerospace industry. Its basic development requires identifying the critical equipment, its critical characteristics and causes of degradation, developing efficient, effective control tasks on that basis, organizing the resulting structured work composed of actionable tasks into a schedule to implement, and then just performing the required outcomes.

In most instances, those will be what is called condition-directed maintenance. What are the benefits of doing something like this? Well, it's going to reduce risk. Why? Because it's far more complete, far more complete. It could prepare for new plant staffing, where we build new plants and we have a lot of transition, people that are not only not familiar with new plants; they're not familiar with nuclear energy in general.

It would be very standardized. Standardized reliability programs would more completely meet the intent of Part 52, the safety intent of Part 52. They would be more consistent. They would actually benefit everyone, because they would remove uncertainty.

A clear reliability assurance program would make projections far more certain. It would lower nuclear costs. You'd know exactly what the scope of work was and the requirements for a specific design. They would highly motivate designers to improve their designs up front, because they would understand, before the plant was built, what the requirements would be and what the costs would be. So it would improve design and operations consistency.

Some of the benefits. Again now, under Part 52, you would have a complete RAP program or complete reliability assurance program, not just a list of equipment. Those would be very measurable, performable activities. They would fulfill the design RAP completely. They would give you the program inputs to current rules, like the maintenance rule.

The maintenance rule right now is a backward-looking rule. It's performance-based, but it looks at performance. It doesn't tell you anything about how to achieve a level of performance. It just measures performance.

If you have the performance requirements going in, you have now a leading indicator. So whereas the maintenance rule is a lagging indicator that's looking back at what happened, if I have something like a comprehensive reliability assurance program and I measure degree of completion of that program, I have a forward-looking indicator. If I complete 50 percent of this program, it tells me what my future expectation will be. If I complete 99 percent of this program, it projects my future expectation.

Why act now? Well, the historical approach is confusing and incomplete. That's what got us to Three Mile Island. The licensee was the developer, not vendor. Who has the expertise? That's really the vendor, okay. Finally, you're going to be able to answer what specific performance-based activities will make SSC reliable?

That is to say you fully address both parts of the RAP, both the design and the operations part. You clarify your responsibilities, designers versus owners. Certified design versus the site-specific components, scopes of each.

You get now the very best program for a certified design. It becomes in essence a part of the design. It's licensed with it. Fundamentally, it's going to support the PRA projections. It's going to be able to deliver your design reliability with far more certainty than something developed later on by the licensee, and there's less overall cost risk.

Why act now? Because it's long overdue. It really is just common sense. It really is common sense. It's consistent; the opportunity is now before we start building new nuclear plants again. In addition, it's a better way of doing things.

It's better methods, not only for safety but for general design. It has very broad benefits. Although I'm arguing safety benefits here, it will have a lot of impact that will help manage costs.

A better question really is to ask why shouldn't we do this? Why not do it? Why wouldn't we want to improve the reliability assurance program, just like we'd improved safety design or quality assurance programs or technical specifications?

If you think for a moment, the reliability assurance program right now that's specified as design RAP is exactly what we were doing in 1979, exactly. Standard nuclear designs really should improve. So how do we address this gap?

Look at a safety analysis, document and share the results. Do a safety evaluation report or multiple reports. Look at the Commission policy and the SECYs that I've got referenced in the back of this presentation. Look at the possibility of doing generic communications, maybe to tell folks that this is what you're contemplating. According to the results, review and revise policy accordingly.

In any event, regardless of what the NRC does, it's very important that industry appreciate that there are minimum requirements and then there is excellence, what INPO is always promoting. Excellence is doing more than the minimum. More than the minimum is providing more than an equipment list as a reliability assurance program.

What should the ACRS do? Take a position. Recommend action. Ask for a response from the staff. Discuss the SECYs. Historically, the ACRS back in the early 90's was very significant in getting the NRC to realize that the operational part of a reliability assurance program at the end of the day had to align with whatever the existing rules were.

More or less the Commission said okay. The operational reliability assurance program simply becomes the maintenance rule, and it made perfect sense. Now the rest of the picture is fill out the D-RAP. Figure out what design reliability assurance is.

Share all these response with the stakeholders, the designers, the potential owners, maybe the folks who might provide some of the assistance, the standards development organizations and so forth, INPO.

Share the conclusions. Be transparent. What are the alternatives? We could ignore this, we could trivialize it, we can claim it's impossible. For many years, we said it's not possible to do this. The technology was not available. We could eliminate it as a requirement. We could say it doesn't really matter.

Or we can preempt it. We could say we've always done this or we're planning to do it or we were going to get around to it. The time just hadn't come yet. The real challenge today is improve safety processes. The real safety issue is to recognize we're never good enough.

We have to allow the designers new options to design plants better. New plants need to take advantage of this one-time opportunity to figure out how to do reliability assurance programs right. It will pay for itself. It's technically feasible; it's very simple to do before construction.

Part 52 needs an effective reliability assurance program. We solved this problem 40 years ago; why don't we use it? 40 years of similar aircraft experience, certifying and using an effective reliability assurance program process. After that, it's just the right thing to do this. Mr. Chairman, that's my presentation. I'll take questions.

CHAIRMAN POWERS: Are there questions?

MEMBER STETKAR: Yes. I'm sorry. Jim, in your paper you draw a lot of analogy to the aerospace industry, and the fact that they've been supplying preventive maintenance programs as part of the certification for an air frame.

How do you answer the question that in the Part 52 process right now, the designer, and I won't mention a specific designer, but the designer specifies requirements for equipment, the functional requirements?

So for example, I need a 6.4 megawatt diesel generator that's able to start within X number of seconds, come up to speed, carry voltage, take a load reject, etcetera. But I don't specify specific manufacturer. I can buy that diesel from any one of probably a dozen different manufacturers. In fact, I don't buy the diesel until after the COL is issued.

So how can at the design stage or the COL stage, how can I specify a preventive maintenance program, which should be specified by the equipment supplier and not the designer. The designer specifies functional requirements. If I buy a Cooper diesel versus a diesel supplied by some other manufacturer, they may have a much different PM program, the same as Boeing or Airbus would have for their air frames.

So how -- I don't understand why we need it? I fully endorse the fact that we need an integrated reliability assurance program, you know, a well-designed preventive maintenance program. It's just a question of where in the licensing stage does that come in?

MR. AUGUST: I think I appreciate what you're saying, John. I think the analogy is that we have, within the combined license, both a high level design and a detail design, and we recognize that the detail design can't be completed until the plant's actually built.

But at a very high level, as soon as we know what types of equipment to require, then we really can specify the functional requirements for that equipment. We can really start to zero in on the types of things that would provide an effective reliability assurance program or scheduled maintenance program, if you will.

By the way, I'd like to clarify. When I use the term "scheduled maintenance," I'm talking in terms of a generic process, which includes operator monitoring in rounds. So it's probably more than what you're thinking of.

But I think there's the same analogy is we don't expect the final as-built plant to be complete when we certify the design. What we do expect is the high level design to be complete. To the same degree we can expect that as soon as we've established functional requirements, we know we're going to have to verify those functional requirements, and that starts us on a path. It starts us on a path.

Certainly, we're not going to be able to complete it until we do the final ITAAC sign-off. But that also illustrates why it's so important to have ITAACs that are going to address this issue, and do so in a detailed way. Not just one ITAAC that verifies that I give a master equipment list; that's ridiculous, right? I have to give more than that.

As an operator, somebody who actually does the work back in the early 80's, one of the things that used to really be a challenge for us was we did not have guidance. We did not have guidance in many instances; we had to create it.

I would have just loved it if, you know, the plants that we were trying to run and operate had had that guidance provided as a part of the license, because it would have removed all the ambiguity that we were facing. John, does that answer your question?

MEMBER STETKAR: That helps, that helps. I'm just trying to get a sense of, you know, where in the process your paper, you know, recognizes at what level of detail, at which stage of the design --

MR. AUGUST: I think you very clearly grasp that there's at least two major levels here, and one is after you've completely specified the functional requirements of the plant, and then there's a later level, after you've actually populated all that, where you have to fill out the details. They're very different.

MEMBER STETKAR: Have you seen, you probably have; I was just reading it last week, the staff's interim staff guidance on the RAP?

MR. AUGUST: I am not sure whether I have or not. I was just talking to somebody earlier, and I have to back and verify that. I think I have, but I can't say that for a fact.

MEMBER STETKAR: I don't know whether it would -- it probably doesn't address things at the level of detail in your paper. It does address, I think anyway in my opinion, some of the gaps in terms of at least providing guidance at least to the staff, and implicitly to applicants, on what criteria need to be satisfied in terms of -- and specifically you mentioned RAP ITAAC. You know, whether it goes far enough, I don't know, but it --

(Simultaneous discussion.)

MR. AUGUST: I'm taking that as an action item. I have to go back and review. I don't know. I'm going to go look for it.

MEMBER STETKAR: It might be worthwhile.

MR. AUGUST: I think I have, but I can't say for a fact.

MEMBER STETKAR: I'm just curious, whether you've --

MR. AUGUST: I've looked at an awful lot of materials. But I can't verify that one. Other questions?

CHAIRMAN POWERS: Well, I'm sitting here struggling a little bit on what you're asking us to do and what the crisis is. We have plants are, core plants that are operating at like a 90 percent capacity factor, who we don't seem to have any what I would call across-the-board maintenance issues that are not being adequately addressed.

We have an experience base that assures us that plant maintenance is plant-specific, that it's going to be difficult to set up a, what you would call a consensus standard template for deciding on what maintenance is going to be, because I can buy John's pump or diesel generator for 15 plants and one of them, presumably, will have different maintenance requirements than the other 14.

So and similarly, we have a Commission that's wanting to move to more risk-informed regulation. That would suggest that we should be using plant-specific information to the extent we can, rather than straight-jacketing the plant with some sort of consensus standard.

So how does that square up? I'm not sure what you're asking us to do --

MR. AUGUST: Well, there's a lot of things that I heard you say, so I'm going to try to answer those one at a time as best I can.

Certainly, having a performance-based or risk-informed approach to regulation is outstanding. However, it's not everything. To get to where I'm performing well, I have to have a plan in advance.

We don't question the fact, for example, that we expect folks to have a quality assurance program, such that when they bring in parts into a plant, we know that there's something behind there.

So we have a lot of these requirements already. Again, the question I'm going to pose to you is really a plant is going to have to have some kind of a very structured reliability assurance program. I'm going to call it a scheduled maintenance plan. Who is responsible for developing it?

Who's responsible for developing it? Is it the certified design supplier? Is it the licensee? Is it both? Is it all of the above with industry. Right now, it's dumped in the lap of the licensee by default. Is that the right solution? I argue that it's not.

They should not be the people who are asked to develop some of these requirements and make these decisions on equipment they don't even understand. Again, for some of you who were around, I mean that's probably every one of you, in the 1980's, you know what I'm talking about. You didn't know what you were supposed to do.

FSAR, safety analysis said this or that, and you really didn't have that much familiarity with the equipment or how best to deal with it and so forth, and so you did the best you could.

Were we as a licensee the best people to figure that out? Heck no, heck no. But was it dumped in our lap? You bet you it was. Would it have been better if some of our suppliers had been forced to answer those questions or at least think about them?

Hey, go back to Three Mile Island and power operated relief valves. Would those things have ever been installed if people had done more to figure out how they were going to operate and maintain them, to make sure they were reliable? My answer is unequivocally not. They would have never installed those things. It would have been unacceptable.

They're just one piece of many, many pieces of equipment that we had back then. Now you're probably thinking, I think I heard you say Dana, "Hey, we've got a lot of experience under the belt now." Okay, great. Do we have any new equipment we're looking at? You betcha. Digital controls. What else? Rotary air compressors.

What else? Mitsubishi wants to put in combustion turbines instead of diesel generators? We don't have much experience there. So how are we going to deal with this? Are we going to go through 30 more years of hit and miss?

My answer, my position is unequivocally not. That's not the way to do this, and you need more structure. You need more structure. Dana, does that answer your question? Have I answered those?

CHAIRMAN POWERS: But still I'm struggling to know what the --

MR. AUGUST: Well, you asked me what do I think the ACRS could do. Let me answer that again. I think the ACRS, in its role as a safety oversight committee, should look at this and basically pass judgment on it. Say yeah, this is a better way to do this, or say no, there's no merit, or say we're not sure.

Why don't we have the staff write up an SER? Why don't we have the staff take a look at it? We don't know. This is the first time we heard about this.

Maybe there's some merit there. We don't know. But I think by putting it with the ACRS, it puts the impetus on the staff to actually look at it, give it a fair shake, go out and look at this other standard.

The reason it's been ignored for 30 years is it's an aerospace standard. Well guess what? Complex equipment that has safety implications, it's the same. It doesn't make any difference. It doesn't make any difference as far as the process is concerned. It could be easily adapted to nuclear applications.

So that's what I think, Dana, that we ought to be doing, is just give it a fair shake and think about it, because I think it's got a tremendous potential to improve the industry, improve safety, improve certainty and reduce cost.

That, by the way, is exactly the aerospace experience. Their maintenance costs dropped substantially when they started doing this about 1970.

CHAIRMAN POWERS: Any other questions?

MEMBER RAY: No questions.

MEMBER RYAN: No questions.

CHAIRMAN POWERS: Thank you.

MR. AUGUST: Thank you very much. I appreciate the opportunity.

CHAIRMAN POWERS: Uh-huh. Okay. Well, I'd say that brings to a close today's session. I guess we're meeting again tomorrow starting at 12:30, to continue on with the --

MEMBER STETKAR: 12:30?

CHAIRMAN POWERS: To continue on with review of the certified design, and transitioning back to the certification at that time, and we'll explore more on the severe accident chapter. Somebody actually thinks we're going to complete that chapter.

On this particular issue, I've taken quite a few notes based on your questions. So if you want to have particular points written, it would be useful. I don't intend to take this forward to the committee at this time, unless there's some pressure from the subcommittee to do so.

My own view was I thoroughly enjoyed today. I thought it was great, even the radiation protection stuff.

MEMBER RYAN: Thank you very much.

CHAIRMAN POWERS: And with that, I will close the meeting for today at this phase and then we'll start again tomorrow.

(Whereupon, at 4:24 p.m., the meeting was recessed, to reconvene on Wednesday, April 21, 2010 at 12:30 p.m.)

UNISTAR NUCLEAR ENERGY

Presentation to ACRS
U.S. EPR^{TM} Subcommittee
Calvert Cliffs Nuclear Power Plant Unit 3
FSAR Chapter 4, Reactor
April 20, 2010

Introduction

- RCOLA authored using ‘Incorporate by Reference’ (IBR) methodology.
- To simplify document presentation and review, only supplemental information, site-specific information, or departures from the U.S. EPR FSAR are contained in the COLA.
- AREVA U.S. EPR FSAR ACRS Meeting for Chapter 4 – Reactor occurred on March 3, 2010.
- Today’s Presentation was prepared by UniStar and is supported by AREVA (U.S. EPR Supplier).
  - Mark Finley (UniStar Engineering Manager)
  - Honqging Xu (AREVA Principal Engineer)

Introduction

- Today Mark Finley, UNE Engineering Manager, will present the Calvert Cliffs Unit 3 FSAR Chapter 4, Reactor.

Chapter 4, Reactor Agenda

- Reactor
  - FSAR content
- NRC SER Open Items
- Conclusions

Chapter 4, Reactor

FSAR Content

- The Calvert Cliffs Unit 3 FSAR Chapter 4 incorporates by reference the U.S. EPR Chapter 4 FSAR.
- The Calvert Cliffs Unit 3 FSAR Chapter 4 requests no departures from the U.S. EPR Chapter 4 FSAR.
- Calvert Cliffs Unit 3 requests an exemption from 10 CFR 50.46 and 10 CFR 50, Appendix K related to the use of M5™ advanced zirconium alloy for the Calvert Cliffs Unit 3 fuel rod cladding and fuel assembly structural material.
- The Calvert Cliffs Unit 3 FSAR Chapter 4 presents no site-specific information in addition to the U.S. EPR Chapter 4 FSAR.
- The U.S. EPR FSAR does not specify any COL information/interface items to be addressed by a COL applicant.

Chapter 4, Reactor

<u>FSAR Content continued</u>

- No ASLB contentions concerning Chapter 4.
- Two NRC SER Open Items.
- No NRC SER Confirmatory Items.

Chapter 4, Reactor
Agenda

- Reactor
  - FSAR content
- NRC SER Open Items
- Conclusions

Chapter 4, Reactor
NRC SER Open Items

NRC SER Open Items

1. RAI 225, Question 04.03-1, (address plant specific surveillance of the reactor internals in regard to fluence methodology benchmarking)
   - UniStar response transmitted by letter UN#10-095, dated April 12, 2010.
2. RAI 226, Question 04.02-1, (add a discussion to Chapter 4 for exemption on the use of M5™ advanced zirconium alloy fuel rod cladding)
   - UniStar response transmitted by letter UN#10-096, dated April 14, 2010.

Chapter 4, Reactor
Agenda

- Reactor
  - FSAR content
- NRC SER Open Items
- Conclusions

Conclusions

- No NRC SER Confirmatory items
- No ASLB Contentions
- Two NRC SER Open items have been received and responses provided.
- The Calvert Cliffs Unit 3 FSAR Chapter 4 requests no departures
- The Calvert Cliffs Unit 3 requests an exemption from 10 CFR 50.46 and 10 CFR 50, Appendix K related to the use of M5™ advanced zirconium alloy for the Calvert Cliffs Unit 3 fuel rod cladding and fuel assembly structural material.

Acronyms

- ACRS-Advisory Committee on Reactor Safeguards
- ACWS-Auxiliary Cooling Water System
- ANSI-American National Standards Institue
- ASLB-Atomic Safety & Licensing Board
- ASME- American Society For Mechanical Engineers
- CDF-Core Damage Frequency
- CFR-Code of Federal Regulations
- COL- Combined License
- COLA-COL Application
- DC-Design Certification
- D-RAP- Design reliability assurance program
- EDF- Électricité de France
- EOP-Emergency Operating Procedures
- EPR- Evolutionary Power Reactor
- FSAR- Final Safety Analysis Report
- FSER – Final Safety Evaluation Report
- IBR-Incorporate by Reference
- IOE -Industry Operating Experience
- JSW-Japan Steel Works
- LRF-Large Release Frequency
- M-Rule-Maintenance Rule
- NEI - Nuclear Energy Institute
- NQA – Nuclear Quality Assurance
- QA – Quality Assurance
- QAPD-quality assurance program description
- Q&PI- Quality & Performance Improvement
- PRA-probabilistic risk analysis
- RAP- Reliability Assurance Program
- RCOLA-Reference COL Application
- SER -safety evaluation report
- SSC-structures, systems, and components
- TR- Topical Report

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

Presentation to the ACRS Subcommittee

Calvert Cliffs Nuclear Power Plant Unit 3, Combined License Application Review

Safety Evaluation Report with Open Items

General Presentation

April 20, 2010

Review Schedule
(Public Milestones)

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

Phase - Activity	**Target Date**
Phase 1 - Preliminary Safety Evaluation Report (SER) and Request for Additional Information (RAI)	April 12, 2010
Phase 2 - SER with Open Items	April 27, 2011
Phase 3 – Advisory Committee on Reactor Safeguards (ACRS) Review of SER with Open Items	July 27, 2011
Phase 4 - Advanced SER with No Open Items	January 31, 2012
Phase 5 - ACRS Review of Advanced SER with No Open Items	May 17, 2012
Phase 6 – Final SER with No Open Items	July 17, 2012

ACRS Phase 3 Review Plan

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

FSAR CHAPTERS GROUPED BY COMPLETION DATES

Group	Chapter(s)	Issue Date	ACRS Meeting
3A-1	8	1/6/2010	2/18/2010
3B-1	4 5 12 17	3/20/2010 3/22/2010 3/12/2010 3/19/2010	4/20/2010 & 4/21/2010
3B-2	10 19	4/20/2010 4/19/2010	5/21/2010
3B3, 3B4, 3B5	Remaining 12 Chapters		Meeting Dates not yet finalized

Information Incorporated by Reference

Several chapters of the COLA FSAR incorporate by reference the U.S. EPR Design Certification application, which is currently being reviewed under Docket No. 52-020.

The staff's review of the COL FSAR for the chapters or sections, which incorporate US EPR FSAR by reference, ensures that the combination of the information incorporated by reference from the U.S. EPR FSAR and the information included in the COL FSAR represents the complete scope of information relating to a specific review topic. A generic RAI 222, Question 01-5, has been issued for tracking the open item pertinent to the concurrent review of the US EPR FSAR.

Generic Open Item:

RAI 222, Question 01-5 tracks the ongoing review of the U.S EPR FSAR as an open item for all COLA chapters. This OI will be closed after the design certification is complete.

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

Presentation to the ACRS Subcommittee

Calvert Cliffs Nuclear Power Plant Unit 3, Combined License Application Review

Safety Evaluation Report with Open Items

Chapter 4: Reactor

April 20, 2010

Staff Review Team

- **Technical Staff**
  - **Fred Forsaty**
    Reactor Systems, Nuclear Performance, and Code Review Branch
  - **John Budzynski**
    Reactor Systems, Nuclear Performance, and Code Review Branch
  - **Shanlai Lu**
    Reactor Systems, Nuclear Performance, and Code Review Branch
  - **John Honcharik**
    Component Integrity Branch
  - **Robert Davis**
    Component Integrity Branch

  - **Project Managers**
  - **Surinder Arora**
  - **Jason Carneal**

Overview of COLA Review

SRP Section/Application Section		Site Specific Yes/No	No. of Questions	Status Number of OI
4.2	Fuel System Design	No	1	1
4.3	Nuclear Design	No	1	1
4.4	Thermal-Hydraulic Design	No	0	0
4.5.1	Control Rod Drive System Structural Materials	No	0	0
4.5.2	Reactor Internals and Core Support Materials	No	0	0
4.6	Functional Design of Reactivity Control Systems	No	0	0
Totals			2	2

Description of Open Items

- RAI 226, Question 04.02-1: Tracks the status of the exemption request for the use of M5™ material identified in Part 7 of the Calvert Cliffs Nuclear Power Plant (CCNPP) Unit 3 combined license application (COLA) and requests a reference to the exemption be added to the FSAR.

- RAI 225, Question 04.03-1: Tracks the need for the combined license (COL) applicant to address the need for a COL information item on fluence methodology benchmarking identified in the safety review of the U.S. EPR design certification application.

Section 4.2 – Fuel System Design

This section of the COLA FSAR incorporates by reference the U.S. EPR Design Certification application. RAI 222, Question 01-5 tracks the ongoing review of the U.S EPR FSAR as an open item.

M5™ Exemption request

The COLA for the Calvert Cliffs Nuclear Power Plant Unit 3, Part 7, Item 1.2.6, includes an exemption on the use of M5™ advanced zirconium alloy fuel rod cladding.

Staff Evaluation

- The FSAR did not contain a reference to the exemption request for M5™ material. The staff is currently reviewing the exemption request.

Open Items

- RAI 226 Question 04.02-1 requests a discussion be added to Chapter 4 of the Calvert COL FSAR, and will be used to track the status of the exemption request

Section 4.3 – Nuclear Design

This section of the COLA FSAR incorporates by reference the U.S. EPR Design Certification application. RAI 222, Question 01-5 tracks the ongoing review of the U.S EPR FSAR as an open item.

Fluence methodology benchmarking

In U.S. EPR RAI 344, Question 04.03-27, AREVA NP Inc. was requested to provide a COL Item in the FSAR for the U.S. EPR to address plant specific surveillance of the reactor vessel in regard to fluence methodology benchmarking. In a letter dated April 12, 2010, UniStar stated that when the COL item is added to the U.S. EPR FSAR, the applicable parts of the COL application for CCNPP Unit 3 would be updated to address this additional requirement.

Staff Evaluation

- The COL FSAR will need to address this COL information item once it is added to the U.S. EPR FSAR.

Open Item

- RAI 225 Question 04.03-1 requests an update to the COL FSAR Tier 2 to include the additional requirements of the COL item.

Conclusion

- The staff's review confirmed that the COL applicant addressed the required information relating to the reactor with the exception of the identified open items.

- The COL applicant is expected to address the outstanding information in the COL FSAR related to this chapter.

- Questions?

Acronyms

United States Nuclear Regulatory Commission
Protecting People and the Environment

- CCNPP – Calvert Cliffs Nuclear Power Plant
- COL – combined license
- COLA – combined license application
- FSAR – Final Safety Analysis Report
- IBR – incorporated by reference
- SER – Safety Evaluation Report
- RAI – request for additional information

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Presentation to ACRS
U.S. EPR™ Subcommittee
Calvert Cliffs Nuclear Power Plant Unit 3
FSAR Chapter 5, Reactor Coolant System
April 20, 2010

Introduction

- RCOLA authored using ‘Incorporate by Reference’ (IBR) methodology.
- To simplify document presentation and review, only supplemental information, site-specific information, or departures from the U.S. EPR FSAR are contained in the COLA.
- AREVA U.S. EPR FSAR ACRS Meeting for Chapter 5 – Reactor Coolant System occurred on March 3, 2010.

Introduction

- No Departures from EPR FSAR for Calvert Cliffs Unit 3, Chapter 5
- No ASLB Contentions
- Six COL Information Items
- Three NRC SER Open Items
- Two NRC SER Confirmatory Items

Introduction

- Today's Presentation was prepared by UniStar and is supported by AREVA (U.S. EPR Supplier).
  - Mark Finley (UniStar Engineering Manager)
  - Dale Matthews (AREVA Supervisory Engineer, Component Design)

Introduction

- Today Mark Finley, UniStar Engineering Manager, will present the Calvert Cliffs Unit 3 FSAR Chapter 5.
- The focus of today's presentation will be on site-specific information that supplements the U.S. EPR FSAR.

Chapter 5, Reactor Coolant System Agenda

- Reactor Coolant System
  - COL Information Items
- NRC SER Open items
- NRC SER Confirmatory items
- Conclusions

Reactor Coolant System
COL Information Items

- Item# 5.2-2
  Identify additional ASME code cases to be used.

The COL Item is addressed as follows:

- ➢ No additional ASME code cases will be utilized for the construction of the Reactor Coolant System components.

Reactor Coolant System
COL Information Items

- Item# 5.2-3

  Identify the implementation milestones for the site-specific ASME Section XI preservice and ISI program for the RCPB, consistent with the requirements of 10 CFR 50.55a(g). The program will identify the applicable edition and addenda of the ASME Section XI, and will identify any additional relief requests and alternatives to Code requirements.

  The COL Item is addressed as follows:

  - Preservice inspection will be implemented prior to initial startup.
  - ISI program will be implemented prior to commercial service.
  - Preservice inspection and ISI programs for the RCPB meet the requirements of 10 CFR 50.55a(g), and comply with ASME Boiler and Pressure Vessel Code, Section XI, 2004 edition.

Reactor Coolant System
COL Information Items

Item# 5.2-3 continued

- The ISI program will incorporate the latest edition and addenda of the ASME Boiler and Pressure Vessel Code Section XI approved in 10 CFR 50.55a(b) on the date 12 months before initial fuel load.
- Inservice examination of components and system pressure tests conducted during successive 120-month inspection intervals will comply with the requirements of the latest edition and addenda of the Code approved in 10 CFR 50.55a(b) 12 months before the start of the 120-month inspection interval (or the optional ASME Code cases listed in Regulatory Guide 1.147, that are defined in 10 CFR 50.55a(b), subject to the limitations and modifications listed in 10 CFR 50.55a(b)).

Reactor Coolant System
COL Information Items

Item# 5.2-3 continued

- Should relief requests be required, they will be developed through the regulatory process and submitted to the NRC for approval in accordance with 10 CFR 50.55a(g)(5). The relief requests shall include appropriate justifications and proposed alternative inspection methods.

Reactor Coolant System
COL Information Items

- Item# 5.3-1

  Identify the implementation milestones for the material surveillance program.

The COL Item is addressed as follows:

- The implementation milestone for the Reactor Vessel material surveillance program is prior to initial fuel load.

Reactor Coolant System
COL Information Items

- Item# 5.3-2

  Provide a plant-specific pressure and temperature limits report (PTLR), consistent with an approved methodology.

The COL Items are addressed as follows:

- ➢ A plant-specific PTLR will be provided in accordance with Calvert Cliffs Unit 3 Technical Specification 5.6.4, “Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR),” and will be based on the methodology provided in ANP-10283P.

Reactor Coolant System
COL Information Items

- Item# 5.3-3

  Provide plant-specific RT_{PTS} (pressurized thermal shock reference temperature) values in accordance with 10 CFR 50.61 for vessel beltline materials.

  The COL Item is addressed as follows:

  - The plant-specific RT_{PTS} values for vessel beltline materials will be determined in accordance with 10 CFR 50.61 and provided to the NRC within one year of acceptance of the reactor vessel by the licensee.

Reactor Coolant System
COL Information Items

- <u>Item# 5.4-1</u>

  Identify the edition and addenda of ASME Section XI applicable to the site-specific Steam Generator inspection program.

  The COL Item is addressed as follows:

  - The Steam Generator Program tube inspections for preservice inspection and the initial ISI interval will comply with ASME Boiler and Pressure Vessel Code, Section XI, 2004 edition. No relief requests or alternatives are required for use of the 2004 Edition of ASME Section XI.

  - The Steam Generator Program tube inspections for the initial ISI interval shall incorporate the latest edition and addenda of the ASME Boiler and Pressure Vessel Code approved in 10 CFR 50.55a(b) on the date 12 months before initial fuel load.

Reactor Coolant System
COL Information Items

Item# 5.4-1 continued

- ISI conducted during successive 120-month inspection intervals will comply with the requirements of the latest edition and addenda of the Code in 10 CFR 50.55a(b) 12 months before the start of the 120-month inspection interval (or the optional ASME Code cases listed in RG 1.147, that are in 10 CFR 50.55a(b), subject to the limitations and modifications listed in 10 CFR 50.55a(b)).

- Should relief requests be required due to the use of code additions/addenda later than the 2004 Edition will be developed through the regulatory process and submitted to the NRC for approval in accordance with 10 CFR 50.55a(g)(5). The relief requests will include appropriate justifications and proposed alternative inspection methods.

Chapter 5, Reactor Coolant System Agenda

- Reactor Coolant System
  - COL Information Items
- NRC SER Open items
- NRC SER Confirmatory items
- Conclusions

Chapter 5, Reactor Coolant System
NRC SER Open Items

- NRC SER Open Items

1. RAI 223, Question 05.02-05-3 (procedures for conversion and alarm setpoints for prolonged low-level unidentified leakage inside containment) is responded to by letter UN#10-093, dated April 14, 2010.
2. RAI 223, Question 05.02-05-4 (specify operator actions in response to prolonged low level leakage conditions that exist above normal leakage rates and below the TS limits) is responded to by letter UN#10-093, dated April 14, 2010.
3. RAI 227, Question 05.04.02.02-13 (revise FSAR Table 13.4-1 to move the Section 5.4.2.5 reference from "Preservice Testing Program" to "Preservice Inspection Program) is responded to by letter UN#10-097, dated April 14, 2010.

Chapter 5, Reactor Coolant System Agenda

- Reactor Coolant System
  - COL Information Items
- NRC SER Open items
- NRC SER Confirmatory items
- Conclusions

Chapter 5, Reactor Coolant System
NRC SER Confirmatory Items

- NRC SER Confirmatory Items

1. Incorporate UniStar response to RAI 186 Question 05.03.02-2 (plant-specific Pressure and Temperature Limits Report prior to initial fuel load) into COL FSAR Section 5.3.2.1 and Part 10 (ITAAC).

2. Incorporate UniStar response to RAI 40 Question 05.04.02.02-1 (FSAR Table 13.4-1 will be updated to include a reference to FSAR Section 5.4.2.5 for the In-service Inspection Program) into COL FSAR Table 13.4-1.

- ➢ UniStar Nuclear Energy will incorporate the RAI responses in Revision 7 of the COLA.

Chapter 5, Reactor Coolant System
Agenda

- Reactor Coolant System
  - COL Information Items
- NRC SER Open items
- NRC SER Confirmatory items
- Conclusions

Conclusions

- COL Information Items, as specified by EPR FSAR, are addressed in Calvert Cliffs Unit 3 FSAR Chapter 5
- No Departures from U.S. EPR FSAR
- No ASLB Contentions
- Three NRC SER Open Items
- Two NRC Confirmatory Items, RAI responses will be incorporated of into Revision 7 of the COLA

Acronyms

- ACRS-Advisory Committee on Reactor Safeguards
- ACWS-Auxiliary Cooling Water System
- ANSI-American National Standards Institue
- ASLB-Atomic Safety & Licensing Board
- ASME- American Society For Mechanical Engineers
- CDF-Core Damage Frequency
- CFR-Code of Federal Regulations
- COL- Combined License
- COLA-COL Application
- DC-Design Certification
- D-RAP- Design reliability assurance program
- EDF- Électricité de France
- EOP-Emergency Operating Procedures
- EPR- Evolutionary Power Reactor
- FSAR- Final Safety Analysis Report
- FSER – Final Safety Evaluation Report
- IBR-Incorporate by Reference
- IOE -Industry Operating Experience
- JSW-Japan Steel Works
- LRF-Large Release Frequency
- M-Rule-Maintenance Rule
- NEI - Nuclear Energy Institute
- NQA – Nuclear Quality Assurance
- QA – Quality Assurance
- QAPD-quality assurance program description
- Q&PI- Quality & Performance Improvement
- PRA-probabilistic risk analysis
- RCOLA-Reference COL Application
- RCPB-Reactor Coolant Pressure Boundary
- SER -safety evaluation report
- SSC-structures, systems, and components
- TR- Topical Report

Presentation to the ACRS Subcommittee

UniStar Calvert Cliffs Nuclear Power Plant (CCNPP) Unit 3
COL Application Review

Safety Evaluation Report

CHAPTER 5: Reactor Coolant System and Connected Systems

April 20, 2010

Order of Presentation

- Surinder Arora – Calvert Cliffs , Lead Project Manager
- Tarun Roy – Chapter 5, Project Manager

Technical Presenters:

- John Wu- Section 5.2.1- Compliance with Codes and Standards
- Tim Steingass- Section 5.2.4- Inservice Inspection of the Reactor Coolant Pressure Boundary
- Chang-Yang Li- Section 5.2.5- Reactor Coolant Pressure Boundary Leakage Detection
- Joel Jenkins-Section 5.3.1- Reactor Vessel Materials
- Steven Downey- Sections 5.3.2 and 5.3.3- P-T limits, Upper-Shelf Energy, and PTS
- Gregory Makar- Sections 5.4.2.1 and 5.4.2.2- Steam Generator Program

Staff Review Team

United States Nuclear Regulatory Commission
Protecting People and the Environment

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- **Shanlai Lu**
  Reactor Systems, Nuclear Performance, and Code Review Branch

- **Project Managers**
- **Surinder Arora**
- **Tarun Roy**

Overview of Calvert Cliffs Chapter 5 Review

SRP Section/Application Section		No. of Questions	Status Number of OI
5.2	Section Title Integrity of the Reactor Coolant Pressure Boundary	7	2
5.3	Section Title Reactor Vessel	2	0
5.4	Section Title Component and Subsystem Design	13	1
Totals		22	3

Description of Open Items:

RAI 223, Question 05.02.05-3: Staff requested that the COL applicant address the procedures for conversion and alarm setpoints to the COL FSAR.

RAI 223, Question 05.02.05-4: Staff requested that the COL applicant provide the leakage detection procedure for prolonged low level leakage.

RAI 227, Question 05.04.02.02-13: Staff requested that the applicant modify COL FSAR Table 13.4-1 to reference COL FSAR Section 5.4.2 under Preservice Inspection Program.

Information Incorporated by Reference

Chapter 5 of the COLA FSAR incorporates by reference the U.S. EPR Design Certification application, which is currently being reviewed under Docket No. 52-020.

Staff Evaluation

- The application stated that these sections of the COL FSAR were incorporated by reference (IBR). The staff reviewed the appropriateness of this information and found it to be acceptable.

  The following sections are IBR.
  5.2.2, 5.2.3, 5.4.1.1, 5.4.7, 5.4.11, 5.4.12, 5.4.13, and 5.4.14

Technical Topics of Interest Section 5.2.1 – Compliance with Codes and Standards

- Calvert Cliffs Unit 3 COL FSAR Section 5.2.1 incorporated by reference U.S. EPR codes and standards in compliance with 10 CFR 50.55a with no departures and supplements.

- COL Information Item 5.2-2:
  The COL applicant should identify additional ASME Code Cases in its COL application for NRC staff review and approval.

- No additional ASME Code Cases are planned to be used for the COL application at this time.

- No Open or Confirmatory Items in SER Section 5.2.1.

Technical Topics of Interest
Section 5.2.4 – Inservice Inspection of the Reactor Coolant Pressure Boundary

- Section 5.2.4 addresses preservice/inservice inspection and testing of Class 1 components and piping
- COL FSAR Section 5.2.4 incorporates by reference US EPR FSAR Tier 2, Section 5.2.4 with no departures
- COL Information Item 5.2-3 requires the applicant identify the ASME Code of record 12 months prior to fuel load, any relief requests, and milestone schedule
- COL applicant stated they will identify the ASME Code of record 12 months prior to fuel load, and construction milestones in accordance with Table 13.4-1
- No open items

Technical Topics of Interest
Section 5.2.5- Reactor Coolant Pressure Boundary Leakage Detection

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- RG 1.45, Revision 1, “Guidance on Monitoring and Responding to Reactor Coolant System Leakage” Regulatory Position C.3, “Operations-Related Positions”
- CCNPP FSAR Section 5.2.5 incorporates by reference of US-EPR FSAR without supplemental information.
- AREVA indicated in its US-EPR RAI responses that the conformance of the above regulatory position relies on the COLA.
- Open Items: The CCNPP3 FSAR needs to provide the supplemental information to address the following:
  - RAI 223, Question 05.02.05-3: The procedures to convert the instrument indications of various leakage detection into common leakage rate.
  - RAI 223, Question 05.02.05-04: The alarm set points and procedures to respond to prolonged low-level Reactor Coolant System (RCS) leakage.
-

- Incorporates U.S. EPR FSAR with no departures or supplements, except for Reactor Vessel (RV) Material Surveillance Program COL Item.

- COL Item for RV Material Surveillance Program.
  COL Applicant identifies implementation milestones in Table 13.4-1 of COL FSAR.

- No Open or Confirmatory Items.

Technical Topics of Interest
Section 5.3.2

P-T Limits, Upper-Shelf Energy, and PTS

- <u>EPR COL Information Item 5.3-2</u>
  - A COL applicant that references the U.S. EPR design certification will provide plant-specific pressure and temperature limits using approved Pressure-Temperature Limits Report (PTLR) methodology

- <u>COL FSAR Tier 2</u>
  - Confirmed use of generic PTLR provided by AREVA
  - Plant specific P/T limits will be submitted prior to fuel load

- <u>Staff Evaluation</u>
  - <u>Staff finds COL Item Acceptable</u> → meets requirements of Appendix G to 10 CFR part 50 and is consistent with approach of GL 96-03

- No Open items

Technical Topics of Interest
Section 5.3.3
P-T Limits, Upper-Shelf Energy, and PTS

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- EPR COL Information Item 5.3-3
  - A COL applicant that references the U.S. EPR design certification to provide plant-specific RT_{PTS} values in accordance with 10 CFR 50.61

- COL FSAR Tier 2
  - Proposed License condition: plant-specific RT_{PTS} values in will be submitted to the NRC within 1 year of acceptance of the reactor vessel

- Staff Evaluation
  - Staff finds COL item acceptable → provides reasonable assurance that requirements of 10 CFR 50.61 will be met

- No Open items

Technical Topics of Interest
Section 5.4.2.1 and 5.4.2.2 – Steam Generator Program

- COL FSAR Section 5.4.2 incorporates by reference U.S. EPR FSAR Tier 2, Section 5.4.2, with no departures
- COL Information Item 5.4-1 directs a COL applicant to identify the edition and addenda of ASME Section XI applicable to the site-specific steam generator (SG) inspection program
- COL applicant stated that the 2004 Edition of the ASME Code would be used for the PSI program and the ASME Code in effect 12 months prior to fuel load would be used for ISI. No relief requests were identified.
- The staff determined that the 2004 edition is incorporated by reference into the NRC regulations (10CFR50.55a) and identified in Section 5.4.2 of the U.S. EPR DCD. The staff also determined that the proposal for ISI complies with 10 CFR 50.55a.
- The COL applicant revised Table 13.4-1 to show that operational programs include inservice inspection of SG tubes
- The COL applicant will revise Table 13.4-1 to show that operational programs include preservice inspection of SG tubes
  - Open Item RAI 227, Question 05.04.02.02-13

Acronyms:

ASME - American Society of Mechanical Engineers
B & PV- ASME Boiler And Pressure Vessel
COL - Combined License
FSAR - Final Safety Analysis Report
GL- Generic Letter
IBR- Incorporate By Reference
PT- Pressure-Temperature
PTLR-Pressure-Temperature Limits Report
RAI - Request For Additional Information
RCS- Reactor Coolant System
RG- Regulatory Guide
RV- Reactor Vessel
SG- Steam Generator

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Presentation to ACRS
U.S. EPR^{TM} Subcommittee
Calvert Cliffs Nuclear Power Plant Unit 3
FSAR Chapter 12, Radiation Protection
April 21, 2010

Introduction

- RCOLA authored using ‘Incorporate by Reference’ (IBR) methodology.
- To simplify document presentation and review, only supplemental information, or site-specific information, or departures from the U.S. EPR FSAR are contained in the COLA.
- AREVA U.S. EPR FSAR ACRS Meeting for Chapter 12 – Radiation Protection occurred on November 19, 2009.

Introduction

- No Departures from U.S. EPR FSAR for Calvert Cliffs Unit 3, Chapter 12
- No ASLB Contentions
- Six COL Information Items
- Four NRC SER Open Items
- Five NRC SER Confirmatory Items

Introduction

- Today's presentation was prepared by UniStar and is supported by AREVA (U.S. EPR Supplier).
  - Tim Kirkham (UniStar Acting Senior Health Physicist)
  - Pedro Perez (AREVA Supervisory Engineer - Radiological Engineering)
  - Dean Hollmann (Bechtel Project Engineer)
  - Gerald McLane (Bechtel Engineering)

Introduction

- Tim Kirkham, UniStar Acting Senior Health Physicist, will present the Calvert Cliffs Unit 3 FSAR Chapter 12, Radiation Protection.
- The focus of today's presentation will be on site-specific information that supplements the U.S. EPR FSAR Chapter 12.

Chapter 12, Radiation Protection Agenda

- Radiation Protection
  - COL Information Items
- NRC SER
  - Open Items
  - Confirmatory Items
- Conclusions

Radiation Protection
COL Information Items

- Item# 12.1-1
  Describe, at a functional level, elements of the ALARA program for ensuring that occupational radiation exposures are ALARA. This program will comply with provisions of 10 CFR Part 20 and be consistent with the guidance in RGs 1.8, 8.2, 8.7, 8.8, 8.9, 8.10, 8.13, 8.15, 8.27, 8.28, 8.29, 8.34, 8.35, 8.36, and 8.38, and the applicable portions of NUREG-1736.

The COL Item is addressed as follows:

➢ Calvert Cliffs Unit 3 incorporates by reference NEI 07-08A, “Generic FSAR Template Guidance for Ensuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)” and NEI 07-03A, “Generic FSAR Template Guidance for Radiation Protection Program Description.”

Radiation Protection
COL Information Items

- Item# 12.2-1

  Provide site-specific information for required radiation sources containing byproduct, source, and special nuclear material that may warrant shielding design considerations. This site-specific information will include a listing of isotope, quantity, form, and use of all sources in this latter category that exceed 100 millicuries.

Radiation Protection
COL Information Items

Item# 12.2-1 continued

The COL Item is addressed as follows:

- The following radiation sources have been identified to be required:

Isotope	Quantity	Form	Geometry	Use	Location
Cf-252	0.5 Ci	Sealed Source	Source Rod	Primary Start-up Source	Reactor Core
Sb-Be	3E+06 Ci	Sealed Source	Source Rod	Secondary Source	Reactor Core
Cs-137	400 Ci	Sealed Source	Special form sealed capsule	Calibration	Elevation 0 feet Access Building
Cs-137	130 mCi	Sealed Source	Special form sealed capsule	Calibration	Elevation 0 feet Access Building
Am-241	0.03 µCi	Sealed Source	Planchet	Calibration	Elevation 0 feet Access Building
AmBe	3 Ci	Sealed Source	Special form sealed capsule	Calibration	Elevation 0 feet Access Building

Radiation Protection
COL Information Items

- Item# 12.3-1

  Provide site-specific information on the extent to which the guidance provided by RG 1.21, 1.97, 8.2, 8.8, and ANSI/HPS-N13.1-1999 is employed in sampling recording and reporting airborne releases of radioactivity.

The COL Item is addressed as follows:

- Procedures detail the criteria and methods for obtaining representative measurement of radiological conditions, including in-plant airborne radioactivity concentrations in accordance with applicable portions of 10 CFR Part 20 and consistent with the guidance in Regulatory Guides 1.21, 1.97, 8.2, 8.8 and 8.10, and ANSI/HPS-N13.1-1999. Additional discussion of radiological surveillance practices is included in the radiation protection program description provided in Section 12.5.

Radiation Protection
COL Information Items

- Item# 12.3-2

  A COL applicant that references the U.S. EPR FSAR will provide site-specific information on estimated annual doses to construction workers in a new unit construction area as a result of radiation from onsite radiation sources from the existing operating plant(s). This information will include bases, models, assumptions, and input parameters associated with these annual doses.

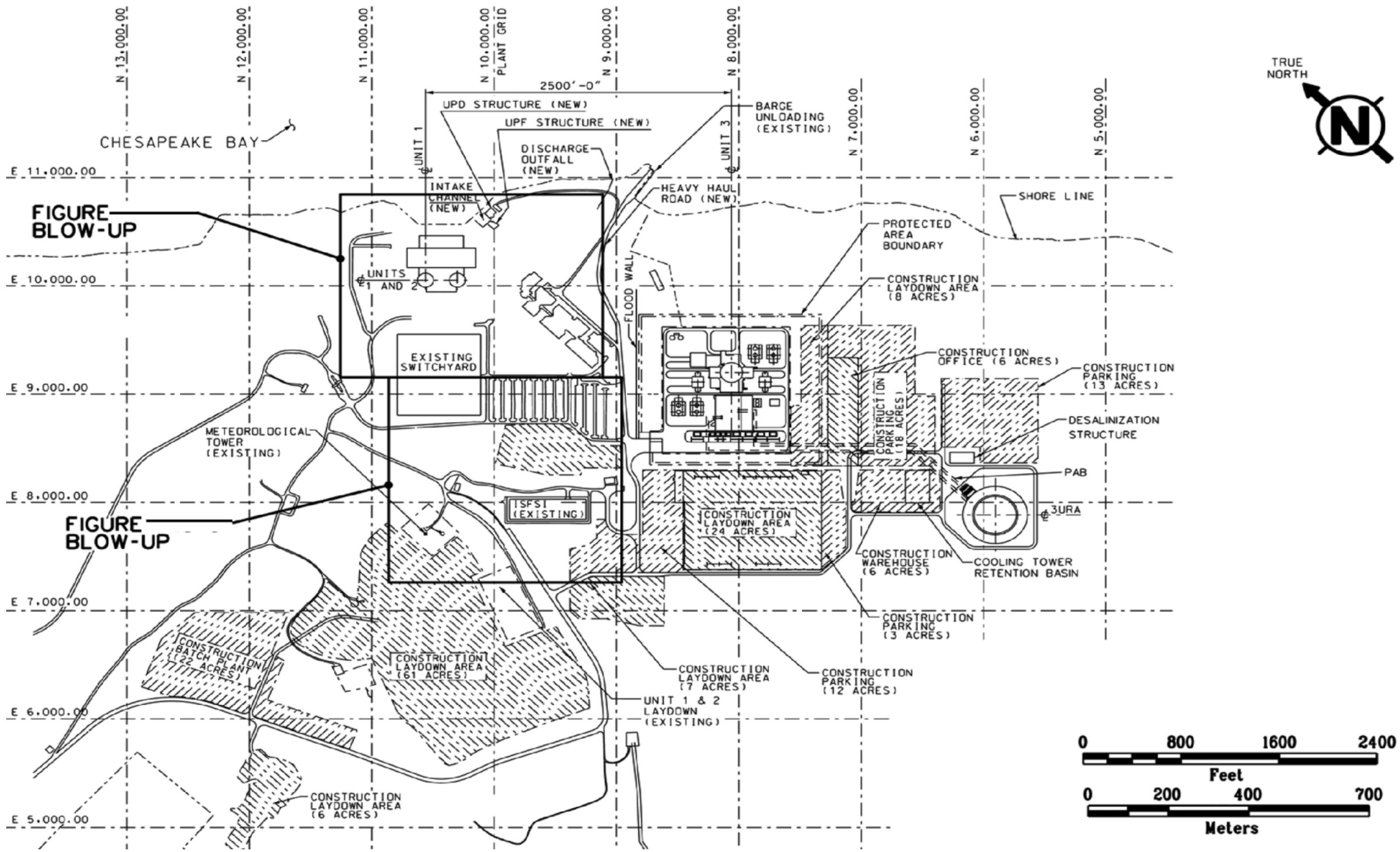
  The COL Item is addressed as follows:

  - Site Layout
  - Radiation Sources at Calvert Units
    - The three main sources of radiation to Calvert Unit 3 workers are gaseous effluents, the Independent Spent Fuel Storage Installation (ISFSI), and the Interim Resin Storage Area.

Radiation Protection
COL Information Items

Item# 12.3-2 continued

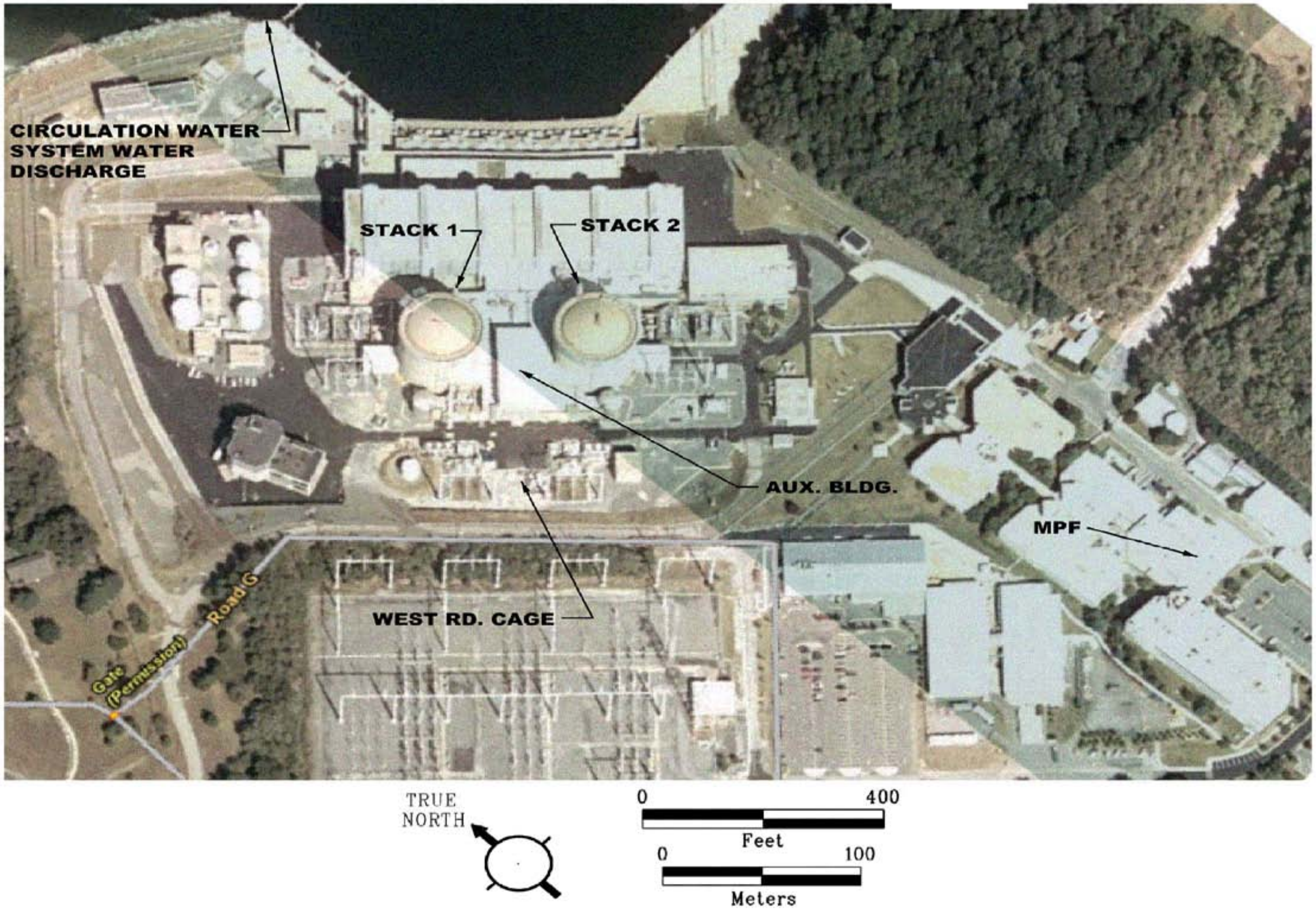
Site Layout of CCNPP Units 1, 2, and 3



Radiation Protection
COL Information Items

Item# 12.3-2 continued

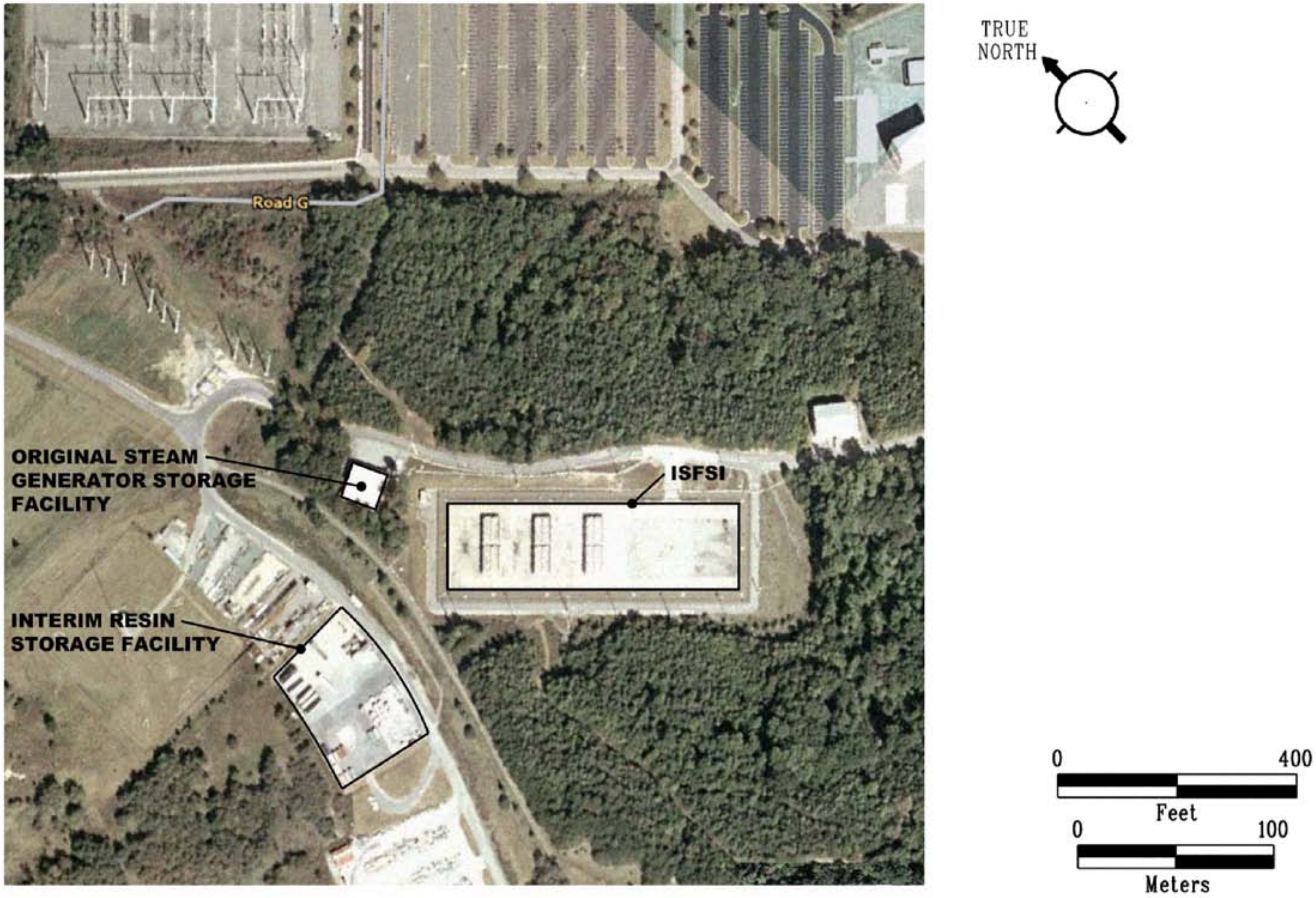
Sources on CCNPP Units 1 and 2 (Part 1 of 2)



Radiation Protection
COL Information Items

Item# 12.3-2 continued

Sources on CCNPP Units 1 and 2 (Part 2 of 2)



Radiation Protection
COL Information Items

Item# 12.3-2 continued

- Basis
  - Sources of radiation from gaseous effluents (Units 1 and 2 Vent stacks) and direct exposure from the ISFSI and the Interim Resin Storage area.
- Models
  - Gaseous dose calculated per the Unit 1 and 2 ODCM and a parametric equation with appropriate χ/Q's used (undepleted, undecayed ground release).
  - Dose rates were calculated per 100' x 100' square on a plant grid from each source.
  - ISFSI source was projected into the future based upon past canister campaigns.

Radiation Protection
COL Information Items

Item# 12.3-2 continued

- Assumptions
  - No change in current effluents
  - ISFSI loading campaigns continue as in past
  - No net change in resin storage
- Inputs
  - TLD data from annual REMP reports for ISFSI and Resin Storage Area
  - Annual TEDE doses from gaseous effluents

Radiation Protection
COL Information Items

Item# 12.3-2 continued

- Result
  - Maximum potential dose to a construction worker is 39 mrem/year if worker spends 2200 working hours in the “road” occupancy zone.
  - Using anticipated occupancy rates and grid method determined highest probable dose to a construction worker.
  - Projected annual dose to a construction worker is less than 3 mrem.

- ALARA
  - Agreement with operating units

Radiation Protection
COL Information Items

- Item# 12.3-3

  Describe the use of portable instruments, and the associated training and procedures, to accurately determine the airborne iodine concentration within the facility where plant personnel may be present during an accident, in accordance with requirements of 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737. The procedures for locating suspected high-activity areas will be described.

The COL Items are addressed as follows:

- A portable monitoring system meeting the NUREG-0737 requirements is available. This monitoring system is incorporated into the emergency plan implementing procedures.
- An in-plant monitoring program includes the training of personnel, procedures for monitoring, and provisions for maintenance of sampling and analysis equipment.

Radiation Protection
COL Information Items

- Item# 12.5-1

  Describe, at the functional level, elements of the Radiation Protection Program. The purpose of the Radiation Protection Program is to maintain occupational and public doses ALARA. The program description will identify how the program is developed, documented, and implemented through plant procedures that address quality requirements commensurate with the scope and extent of licensed activities. This program will comply with the provisions of 10 CFR Parts 19, 20, 50, 52, and 71 and be consistent with the guidance in RGs 1.206, 1.8, 8.2, 8.4, 8.5, 8.6, 8.7, 8.8, 8.9, 8.10, 8.13, 8.15, 8.27, 8.28, 8.29, 8.34, 8.35, 8.36, and 8.38, and the consolidated guidance in NUREG-1736.

Radiation Protection
COL Information Items

Item# 12.5-1 continued

The COL Item is addressed as follows:

- Calvert Cliffs Unit 3 incorporates by reference NEI 07-03A, “Generic FSAR Template Guidance for Radiation Protection Program Description” with the following supplemental information:
  - NEI 07-03A Section 12.5.4.4, Access Control

    The Very High Radiation Areas (VHRAs) located in the Reactor and Fuel Buildings; their locations are shown in U.S. EPR FSAR. VHRAs that are accessible, will be controlled via physical barriers and positive access control, such as VHRA keys that are maintained under the control of the Radiation Protection Manager.
  - NEI 07-03A Section 12.5.4.12, Quality Assurance

    The Quality Assurance program is described in FSAR Section 17.5.

Chapter 12, Radiation Protection Agenda

- Radiation Protection
  - COL Information Items
  - U.S.EPR NRC SER Discussion Items
- NRC SER
  - Open Items
  - Confirmatory Items
- Conclusions

Radiation Protection
NRC SER Open Items

- NRC SER Open Items
  1. RAI 199, Question 12.03-12.04-6 (ALARA program for construction workers)
  2. RAI 199, Question 12.03-12.04-7 (Dose value conversion)
  3. RAI 199, Question 12.03-12.04-8 (Vacuum Breaker Releases)
  4. RAI 199, Question 12.03-12.04-9 (Buried Pipe)

The NRC SER Open Items are addressed as follows:

- ➢ RAI 199, Questions 12.03-12.04 -6, -7, -8, and -9, responses were transmitted by UniStar letter UN#10-020, dated February 24, 2010.

Radiation Protection
NRC SER Confirmatory Items

- NRC SER Confirmatory Items
  1. Incorporate UniStar response to RAI 147 Question 12.01-4 (NEI 07-08A and NEI 07-03A) into COL FSAR Section 12.1, 12.5, Table 1.6-1 and Part 11D.
  2. Incorporate UniStar response to RAI 147, Question 12.01-5 (Table 13.4-1) into COL FSAR Table 13.4-1.
  3. Incorporate UniStar response to RAI 44, Question 12.03-12.04-1 (NEI 08-08A and incorporate 10 CFR 20.1406 into Table 13.4-1) into COL FSAR Section 12.3.
  4. Incorporate UniStar response to RAI 176, Question 12.03-12.04-5 (Access Building conceptual design) into COL FSAR Section 12.3.1
  5. Incorporate UniStar response to RAI 53, Question 12.05-1 (Access to VHRA & Quality Assurance program) into COL FSAR Section 12.5.

The NRC SER Confirmatory Items are addressed as follows:

- ➢ UniStar Nuclear Energy will incorporate the Chapter 12 RAI responses into Revision 7 of the COLA.

Chapter 12, Radiation Protection Agenda

- Radiation Protection
  - COL Information Items
  - U.S.EPR NRC SER Discussion Items
- NRC SER
  - Open Items
  - Confirmatory Items
- Conclusions

Conclusions

- COL Information Items, as specified by U.S. EPR FSAR, are addressed in Calvert Cliffs Unit 3 FSAR Chapter 12
- No Departures from U.S. EPR FSAR for Chapter 12 of the Calvert Cliffs Unit 3 COL
- No ASLB Contentions
- Four NRC SER Open Items – responses provided
- Five NRC Confirmatory Items (Incorporation of RAI responses in Revision 7 of the COLA)

Acronyms

- ACRS-Advisory Committee on Reactor Safeguards
- ANSI-American National Standards Institue
- ALARA- as low as reasonable achieveable
- ASLB-Atomic Safety & Licensing Board
- ASME- American Society For Mechanical Engineers
- CDF-Core Damage Frequency
- CFR-Code of Federal Regulations
- COL- Combined License
- COLA-COL Application
- DC-Design Certification
- D-RAP- Design reliability assurance program
- EDF- Électricité de France
- EOP-Emergency Operating Procedures
- EPR- Evolutionary Power Reactor
- FSAR- Final Safety Analysis Report
- FSER – Final Safety Evaluation Report
- IBR-Incorporate by Reference
- IOE -Industry Operating Experience
- LRF-Large Release Frequency
- M-Rule-Maintenance Rule
- NEI - Nuclear Energy Institute
- NUREG-US Nuclear Regulatory Commission Regulation
- NQA – Nuclear Quality Assurance
- QA – Quality Assurance
- QAPD-quality assurance program description
- RCOLA-Reference COL Application
- REMP-Radiological Environmental Monitoring Program
- SER -safety evaluation report
- SSC-structures, systems, and components
- TEDE-Total Effective Dose Equivalent
- TR- Topical Report

United States Nuclear Regulatory Commission

Protecting People and the Environment

Presentation to the ACRS Subcommittee

Calvert Cliffs Unit 3 Combined License Application Review

SER/OI Chapter 12

Radiation Protection

April 21, 2010

Staff Review Team

- **Technical Staff**
  - **Sara Bernal**
    Health Physics Branch

- **Project Managers**
  - **Surinder Arora**
  - **Jason Jennings**

Overview of COLA

SRP Section/Application Section		Site Specific Yes/No	No. of Questions	Status Number of OI
12.1	Ensuring that Occupational Radiation Exposures are ALARA.	Yes	4	0
12.2	Radiation Sources	Yes	2	0
12.3-12.4	Radiation Protection Design Features	Yes	5	4
12.5	Operational Radiation Protection Program	Yes	1	0
Totals			12	4

Description of Open Items

- OI # 199 Question 12.03-12.04-6: Applicant was asked to describe the ALARA program that will be used to monitor the dose to construction workers.
- OI # 199 Question 12.03-12.04-7: Staff requested that the applicant correct dose unit conversion errors between rem and Sievert.
- OI # 199 Question 12.03-12.04-9: Applicant was asked to identify any site specific buried piping that could potentially become contaminated and describe design features or monitoring that will be credited for demonstrating compliance with 10 CFR 20.1406.
- OI # 199, Question 12.03-12.04-8: The applicant was asked to described design features and/or monitoring for vacuum breakers, if relevant, to demonstrate compliance with 10 CFR 20.1406.

Technical Topics of Interest

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

- 12.1 – Ensuring that Occupational Exposures are ALARA
  - **COL Information Item 12.1-1, ALARA Program**, functional description of applicant's ALARA program
    - Applicant incorporates by reference NEI 07-08, revision 3, "Generic FSAR Template Guidance for Ensuring that Occupational Exposures are ALARA" and NEI 07-03, "Generic FSAR Template Guidance for Radiation Protection Program Description," revision 7.
    - Staff is tracking as a confirmatory item the revision of FSAR Table 13.4-1 to include a reference to section 12.1 of the FSAR for the radiation protection program.
    - Staff is tracking as a confirmatory item the incorporation by reference of the NRC accepted version of NEI 07-08 (NEI 07-08A, revision 0) and NEI 07-03 (NEI 07-03A, revision 0).

Technical Topics of Interest

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

- 12.2 – Radiation Sources
  - **COL Information Item 12.2-1, Site Specific Sources**, Describe byproduct, source, and SNM radiation sources above 100 mCi.
    - Applicant provided a table of radiation sources to be used for start up and calibration.
    - In response to staff's question on how neutron instrumentation would be calibrated, the applicant added a neutron calibration source in FSAR subsection 12.2.1.13 (RAI 108, Question 12.02-1).
    - In response to RAI 119, Question 12.02-2 the applicant described how the National Source Tracking requirements of 10 CFR 20.2207 and the security requirements of 20.1801 would be met.

Technical Topics of Interest

- 12.3-12.4 – Radiation Protection Design Features
  - **COL Information Item 12.3-1, Regulatory Guidance**, Describe application of RGs 1.21, 1.97, 8.2, 8.8, and ANSI N13.1-1999 to sampling, recording and reporting airborne releases of radioactivity.
    - FSAR states that applicant will adhere to above guidance by using it to develop procedural criteria and methods for obtaining representative samples of radioactive material, including airborne radioactivity.
    - Applicant's FSAR also describes types and frequencies of surveys as well as reporting requirements.

Technical Topics of Interest

- 12.3-12.4 – Radiation Protection Design Features
  - **COL Information Item 12.3-2, Dose to construction workers**. Site specific information requested on estimated annual doses to construction workers due to operating Units 1 & 2
    - Dose to construction workers assessment in the FSAR included dose from Units 1 and 2 effluents and direct radiation.
    - Staff requested applicant describe the ALARA program for the construction workers. (**Open Item 199 Question 12.03-12.04-6**).
    - Staff requested correction of dose unit conversion errors in the FSAR (**Open Item 199, Question 12.03-12.04-7**)

Technical Topics of Interest

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

- 12.3-12.4 – Radiation Protection Design Features
  - **COL Information Item 12.3-3, Determination of Post Accident Airborne Iodine Concentrations.** Description of instrumentation, training, and maintenance associated with measuring post-accident iodine concentrations in compliance with 10CFR50.34(f)(2)(xxvii) and criteria of NUREG 0737.
    - FSAR states that the use of portable instrumentation will be incorporated into emergency procedures, while in-plant radiation monitoring program will train personnel, implement procedures for monitoring and maintain the instrumentation in conformance with RG 1.21, RG 8.8 and NUREG-0737 Item III.D.3.3.

Technical Topics of Interest

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

- 12.3-12.4 – Radiation Protection Design Features
  - Compliance with 10 CFR 20.1406, Minimization of Contamination
    - Staff requested site-specific design features that demonstrate compliance with 10 CFR 20.1406, including any applicable to potentially contaminated buried piping, and vacuum breakers (**Open Items 199, Question 12.03-12.04-9 and Question 12.03-12.04-8**)
  - Access Building Conceptual design information
    - Staff is tracking as a confirmatory item the incorporation by reference of the conceptual design information contained in the EPR design certification related to the personnel decontamination area, portable instrument calibration facility, respiratory facility, equipment decontamination facility, radioactive materials storage area, and facility for dosimetry and bioassay

Technical Topics of Interest

- 12.5 – Radiation Protection Program
  - **COL Information Item 12.05-1, Operational Radiation Protection Program Description**, functional description of the Operational Radiation protection Program.
    - Applicant incorporates by reference NEI 07-03, revision 7, “Generic FSAR Template Guidance for Radiation Protection Program Description to address this COL item.
    - Staff is tracking as a confirmatory item the incorporation by reference of the NRC accepted version of NEI 07-03 (NEI 07-03A, revision 0).
    - Site specific information related to very high radiation areas and quality assurance.

Conclusion

- Due to Open Items and Confirmatory Items, the staff is unable to finalize conclusions concerning Chapter 12, “Radiation Protection” at this time.

- Questions?

UNISTAR NUCLEAR ENERGY

Presentation to ACRS
U.S. EPR^{TM} Subcommittee
Calvert Cliffs Nuclear Power Plant Unit 3
FSAR Chapter 17
Quality Assurance and Reliability Assurance
April 20, 2010

Introduction

- RCOLA authored using ‘Incorporate by Reference’ (IBR) methodology.
- To simplify document presentation and review, only supplemental information, or site-specific information, or departures from the U.S. EPR FSAR are contained in the COLA.
- AREVA U.S. EPR FSAR ACRS Meeting for Chapter 17 – Quality Assurance and Reliability Assurance occurred February 18, 2010.

Introduction

- No Departures from EPR FSAR for Calvert Cliffs Unit 3, Chapter 17
- No ASLB Contentions
- Twelve COL Information Items
- Ten NRC SER Open Items
- Two NRC SER Confirmatory Items

Introduction

- Today's Presentation was prepared by UniStar and is supported by AREVA (U.S. EPR Supplier).
  - Mark Harvey (UniStar Director of Quality & Performance Improvement)
  - Richard Szoch (UniStar Director of Testing & Programs Development)
  - Gene Hughes (UniStar Acting Director of PRA)
  - Charles Tally (AREVA Manager Engineering Integration - New Plants)
  - Josh Reinert (AREVA COLA PRA Lead)
- The focus of today's presentation will be on site-specific information that supplements the U.S. EPR FSAR.

Quality Assurance and Reliability Assurance Agenda

- Quality Assurance
  - COL Information Item
  - Oversight Activities
  - NRC SER Open Item
  - NRC SER Confirmatory Item
- Reliability Assurance
  - COL Information Items
  - NRC SER Open Items
- Maintenance Rule
  - COL Information Items
  - NRC SER Open Items
  - NRC SER Confirmatory Item
- Conclusions

Chapter 17, Quality Assurance and Reliability Assurance
Subsections: 17.1, 17.2, 17.3, 17.5
Quality Assurance

Presented by Mark Harvey
UniStar Director of Quality & Performance Improvement

Quality Assurance
FSAR Content

- Section 17.1, "Quality Assurance During Design"
  This section is incorporated by reference.

- Section 17.2, "Quality Assurance During the Operations Phase"
  The Quality Assurance Program is provided in Section 17.5 for the Construction and Operations Phase.

- Section 17.3, "Quality Assurance Program Description"
  This section is incorporated by reference.

- Section 17.5, "Quality Assurance Program Description"
  Program established and approved by the NRC.

Quality Assurance
COL Information Item

- Item# 17.2-1
  Describe the Quality Assurance Programs associated with the construction and operations phases.

The COL Item is addressed as follows:

- ➢ The UniStar QA Program is stated in UniStar Topical Report No. UN-TR-06-001-A, “Quality Assurance Program Description” and conforms to the criteria established in 10CFR 50, Appendix B and commits to implement:
  - Basic Requirements and Supplements of ANSI/NQA-1-1994, “Quality Assurance Requirements for Nuclear Facility Applications,” as described in the QAPD.
  - Specific subparts of NQA-1-1994, as described in the QAPD.

Quality Assurance
Quality Oversight Activities

- Quality & Performance Improvement (Q&PI) Oversight Activities
  - Qualified Suppliers with full scope audit (AREVA, Bechtel, EDF/Ceidre, etc)
  - Qualified EDF/Ceidre by full scope audit to perform field oversight of Creusot Forge and JSW for heavy forgings
  - Reports of oversight activities completed every month
  - UniStar Q&PI to assess EDF/Ceidre oversight at JSW in May 2010
  - Large component oversight – Q&PI experience includes RPV forging and fabrication

Quality Assurance
NRC SER Open Item

- NRC SER Open Item

1. RAI 200, Question 17.05-6 (address each of the regulatory positions in RG 1.33).

The NRC SER Open Item is addressed as follows:

➢ Response letter UN#10-106 is in preparation.

Quality Assurance
NRC SER Confirmatory Item

- NRC SER Confirmatory Item

  1. Incorporate UniStar response to RAI 120, Question 17.05-3 (remove redundant requirements of 10 CFR 50.55(e)) from the QAPD, Topical Report No. UN-TR-06-001-A.

  The NRC SER Confirmatory Item is addressed as follows:

  - UniStar Nuclear Energy will revise the QAPD to address RAI 120 response in the next revision.

Quality Assurance and Reliability Assurance Agenda

- Quality Assurance
  - COL Information Item
  - Oversight Activities
  - NRC SER Open Item
  - NRC SER Confirmatory Item
- Reliability Assurance
  - COL Information Items
  - NRC SER Open Items
- Maintenance Rule
  - COL Information Items
  - NRC SER Open Items
  - NRC SER Confirmatory Item
- Conclusions

Chapter 17, Quality Assurance and Reliability Assurance
Subsections: 17.4,
Reliability Assurance Program

Presented by Richard Szoch
UniStar Director of Testing & Programs Development
and
Gene Hughes,
UniStar Acting Director of PRA

Reliability Assurance
COL Information Items

- <u>Item# 17.4-1</u>

  Identify the site-specific SSCs within the scope of the RAP.

  The COL Item is addressed in COLA FSAR Section 17.4.2 as follows:

  - UniStar response to RAI No. 61, Question 17.04-1, acknowledged AREVA U.S. EPR FSAR listing of the U.S. EPR SSCs within the scope of the RAP. In conjunction, the UniStar response provided the new FSAR Table 17.4-1, Site Specific Systems and Structures.

    - U.S. EPR FSAR Tier 2 Tables 17.4-1 and 17.4-2 specify the U.S. EPR FSAR SSCs that are included within the scope of RAP. These SSCs are incorporated by reference into the Calvert Cliffs Unit 3 RAP.

    - Calvert Cliffs Unit 3 COLA Table 17.4-1 provides the site-specific list of systems and structures within the scope of the RAP.

Reliability Assurance
COL Information Items

- <u>Item# 17.4-2</u>

  Provide the information requested in Regulatory Guide 1.206, Section C.I.17.4.4.

  The COL Item is addressed in COLA FSAR Section 17.4.4 as follows:

  - Reliability assurance activities are implemented in two stages.
    - Stage 1 encompasses D-RAP including procurement, construction, and fabrication and testing leading up to initial fuel load.
    - Stage 2 reliability assurance activities are conducted principally by Calvert Cliffs 3 Nuclear Project, LLC and UniStar Nuclear Operating Services, LLC and commence during the transition to fuel load and plant operation. They are implemented as part of the Maintenance Rule (MR) program as well as other key programs. Stage 2 reliability assurance activities continue for the life of the plant.

Reliability Assurance
COL Information Items

Item# 17.4-2 continued -- RAP Process

RAP

EXPERT PANEL REVIEW

RISK SIGN'T SSC LIST

- CAP/QA
- Procurement
- Fabrication
- Construction
- Installation
- Testing

M-RULE
NEI 07-02A
Program

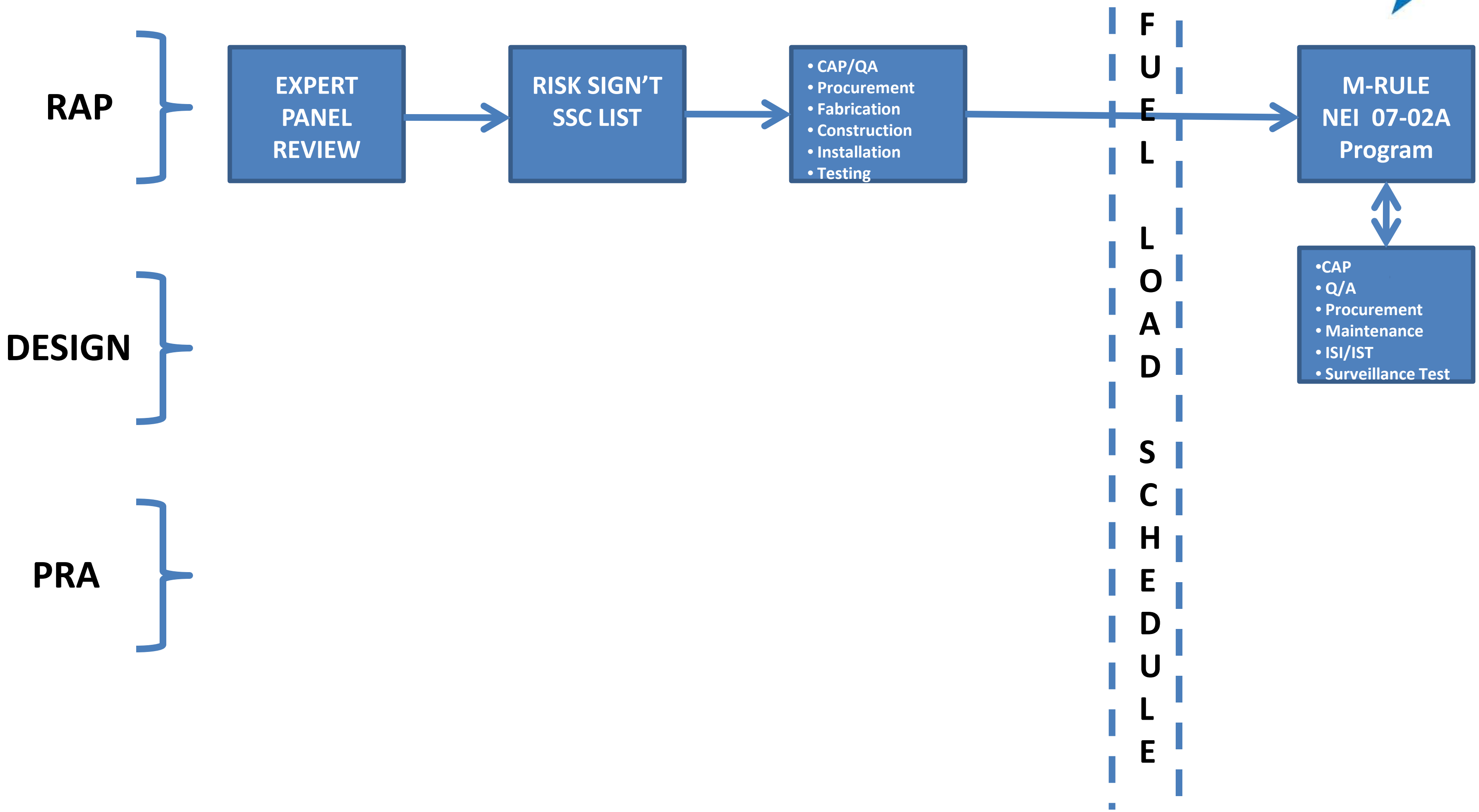
DESIGN

- CAP
- Q/A
- Procurement
- Maintenance
- ISI/IST
- Surveillance Test

PRA

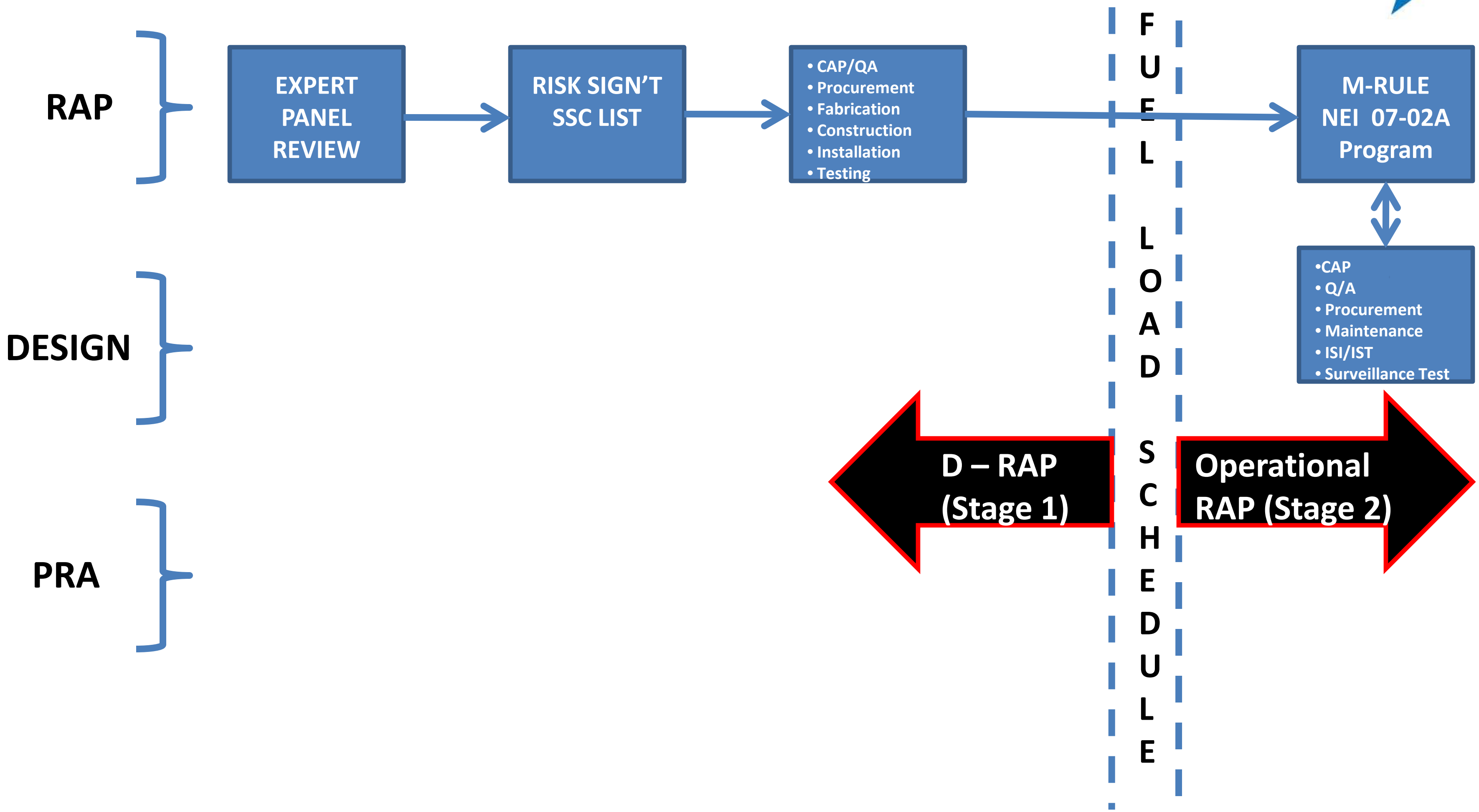
Reliability Assurance
COL Information Items

Item# 17.4-2 continued -- RAP Process



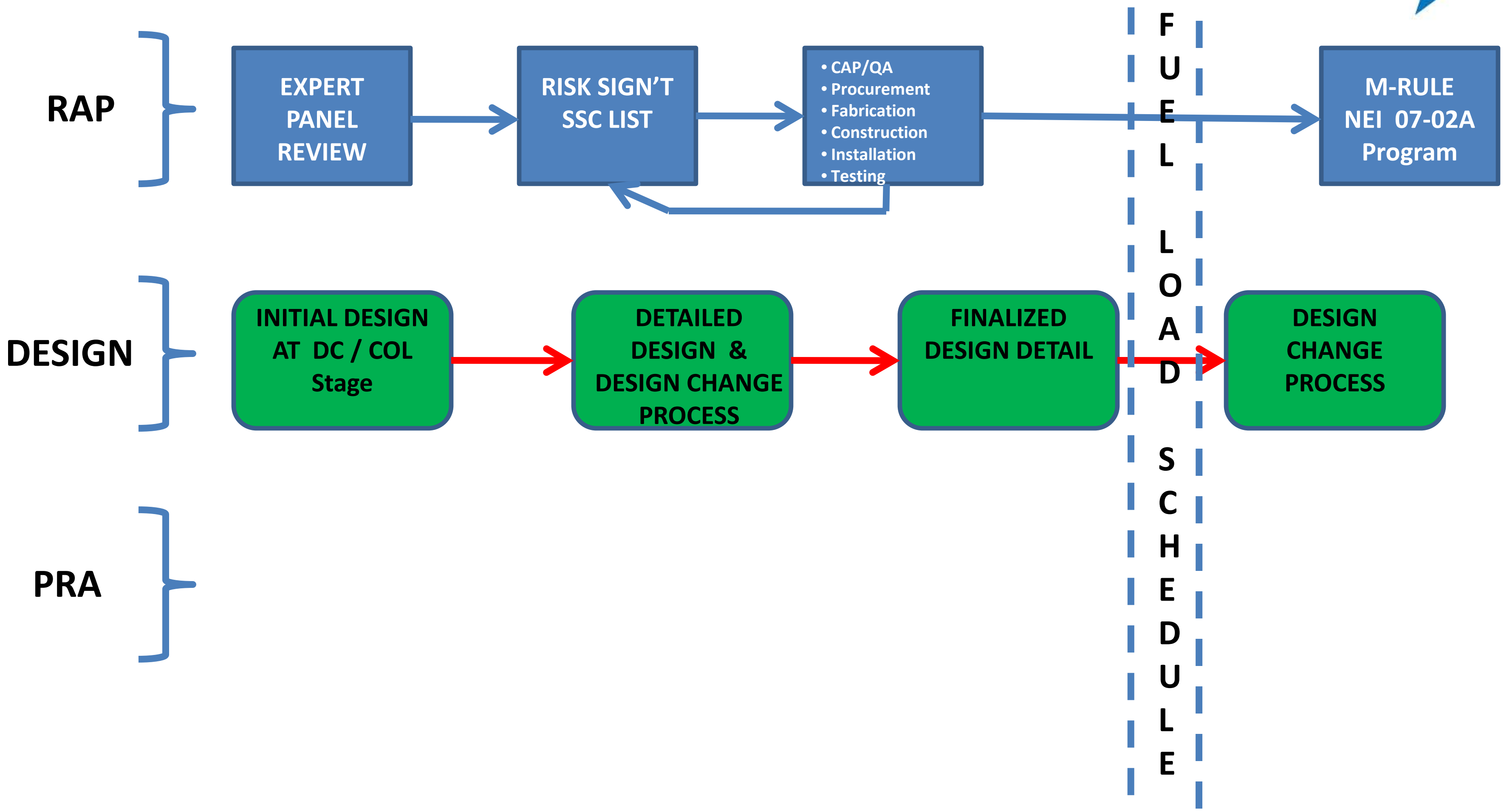
Reliability Assurance
COL Information Items

Item# 17.4-2 continued -- RAP Process



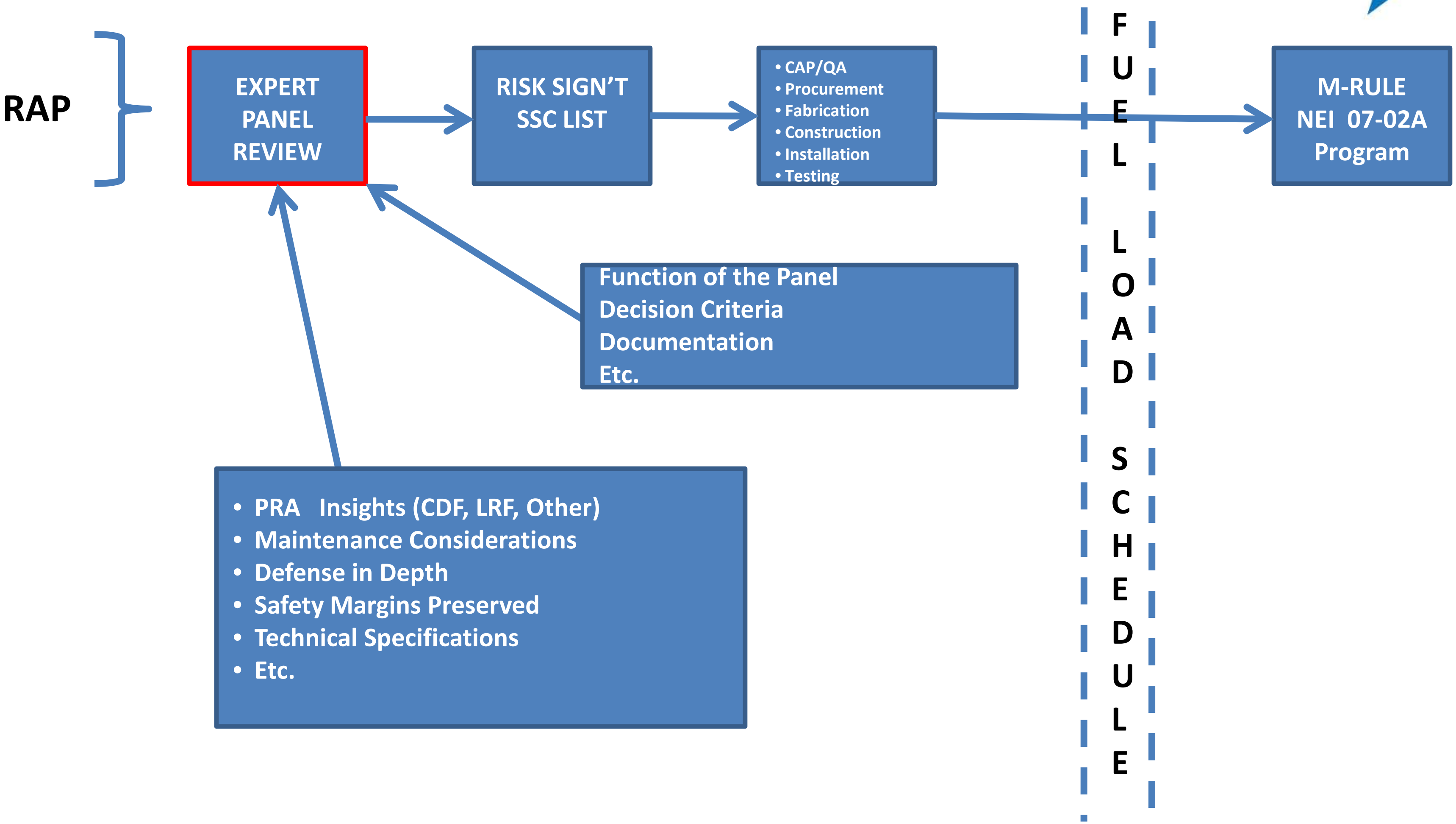
Reliability Assurance
COL Information Items

Item# 17.4-2 continued -- Design Process



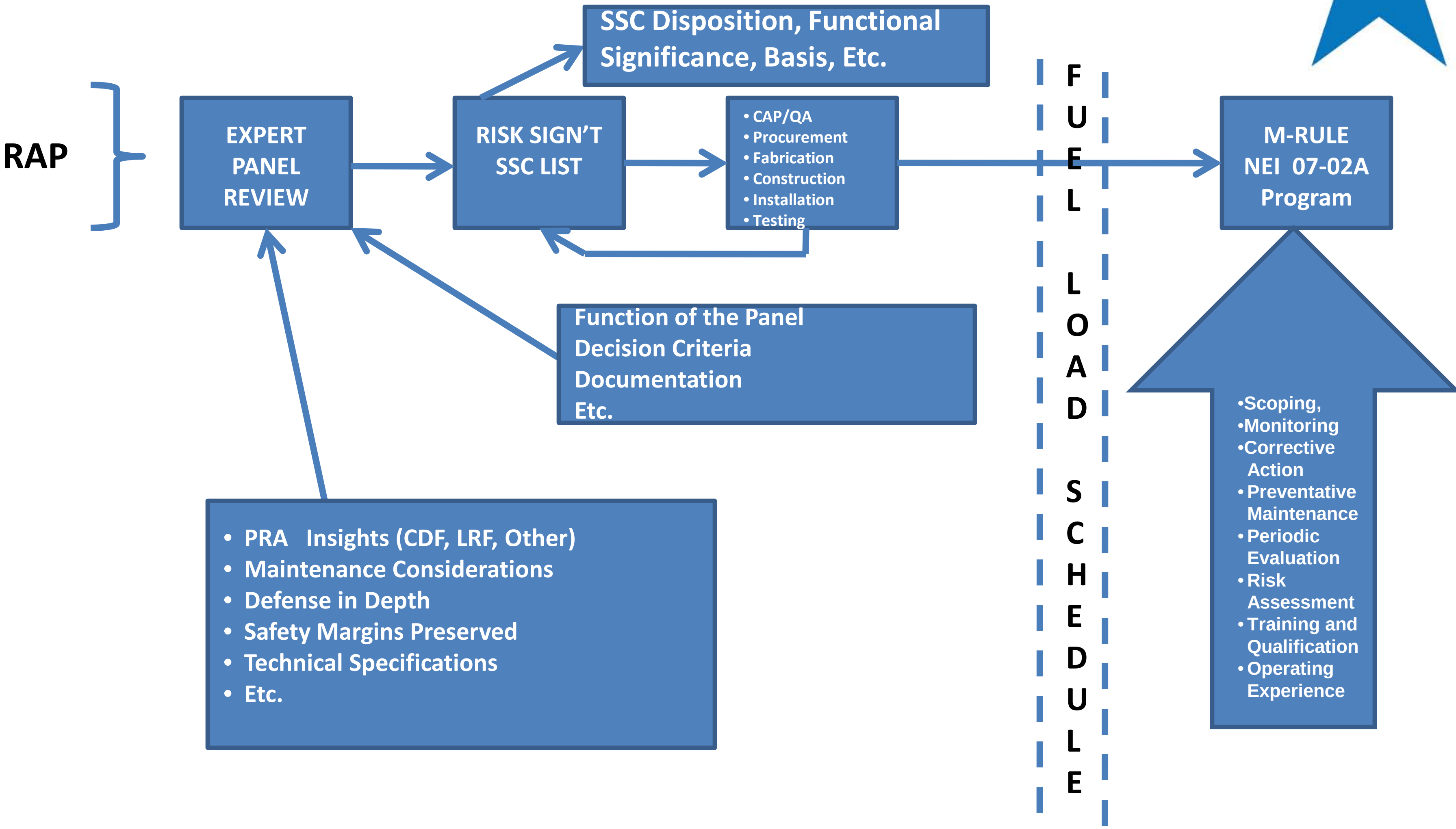
Reliability Assurance
COL Information Items

Item# 17.4-2 continued -- Expert Panel/SSC



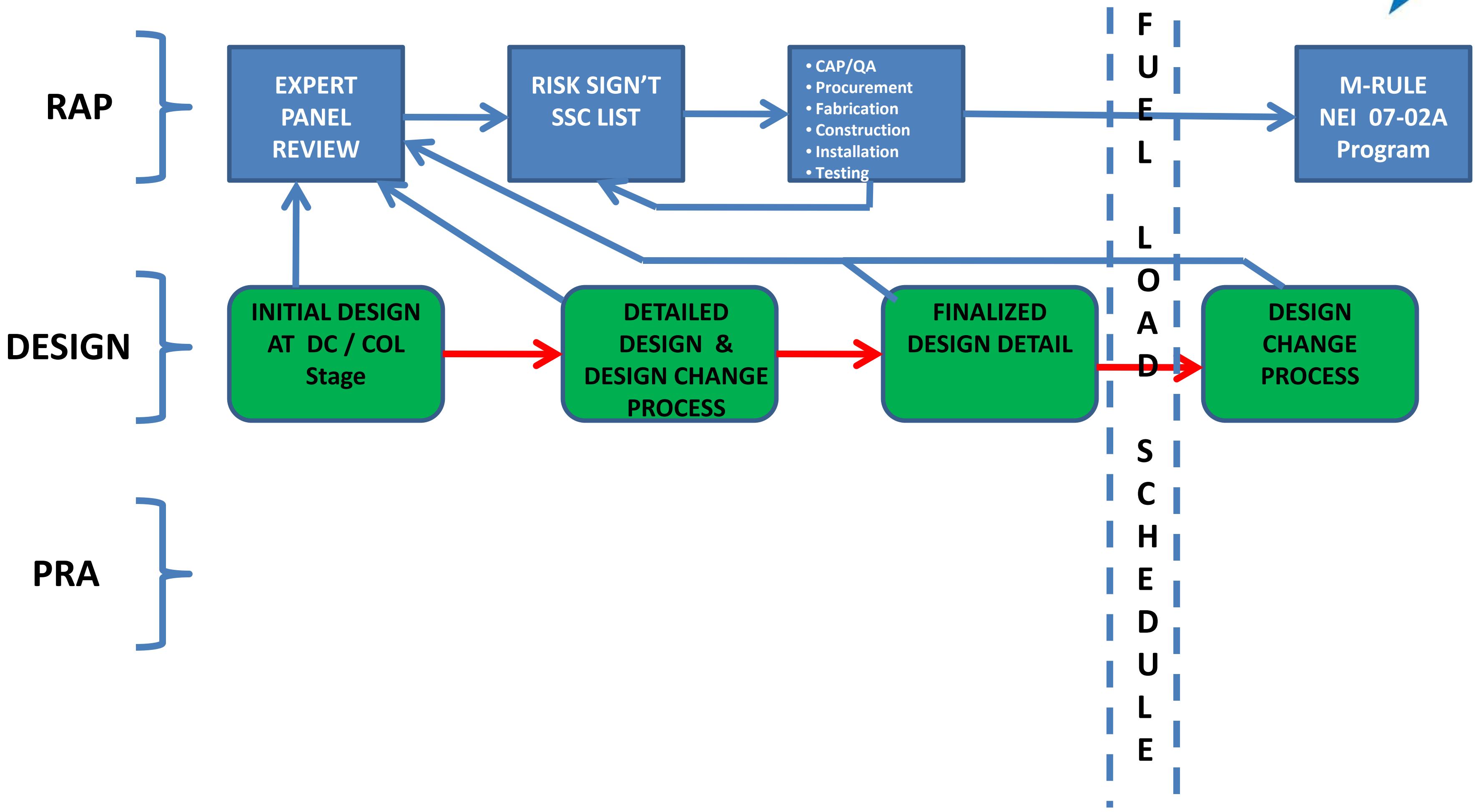
Reliability Assurance
COL Information Items

Item# 17.4-2 continued -- Expert Panel/SSC



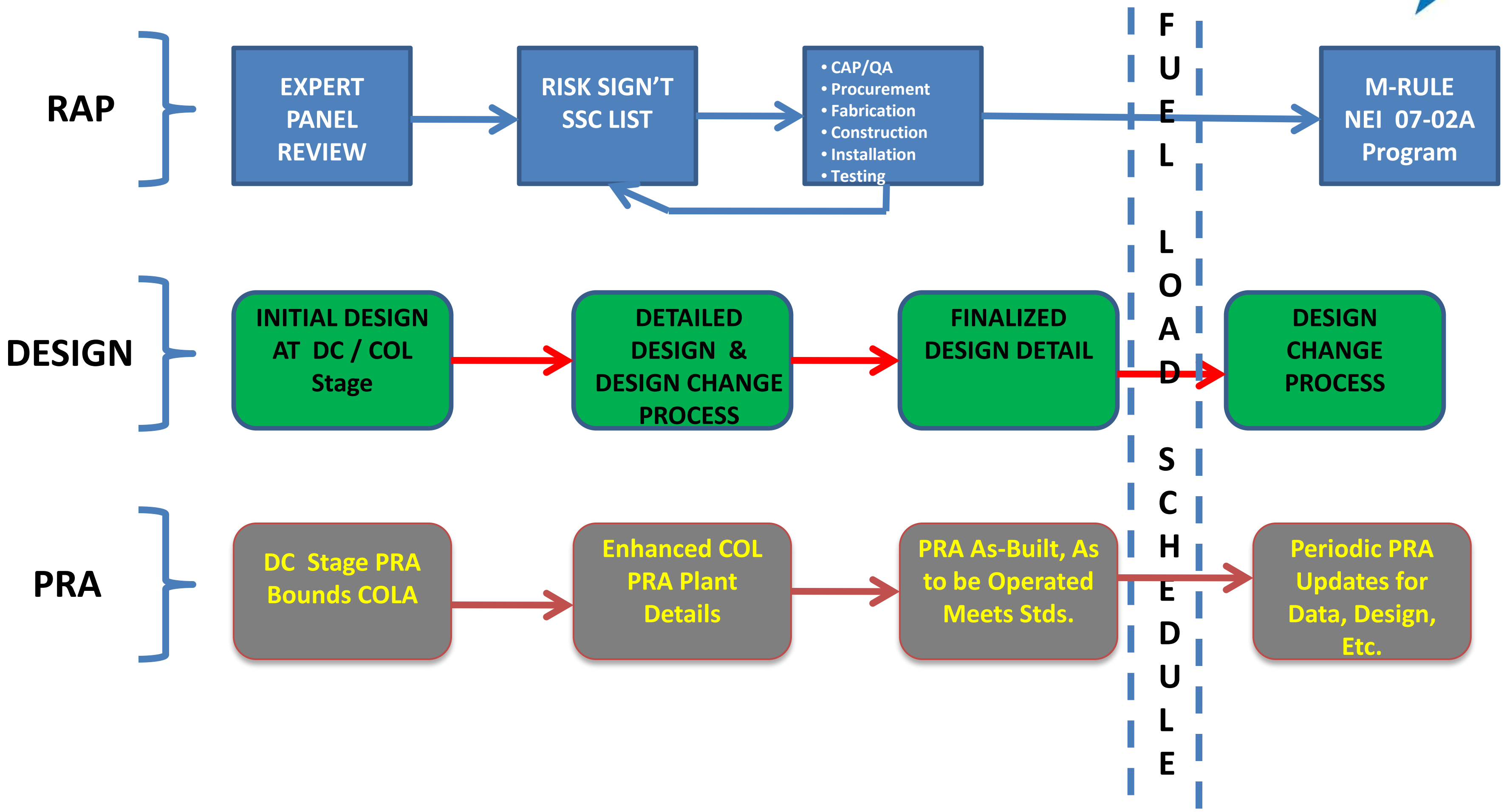
Reliability Assurance
COL Information Items

Item# 17.4-2 continued -- Design/Expert Panel Interface



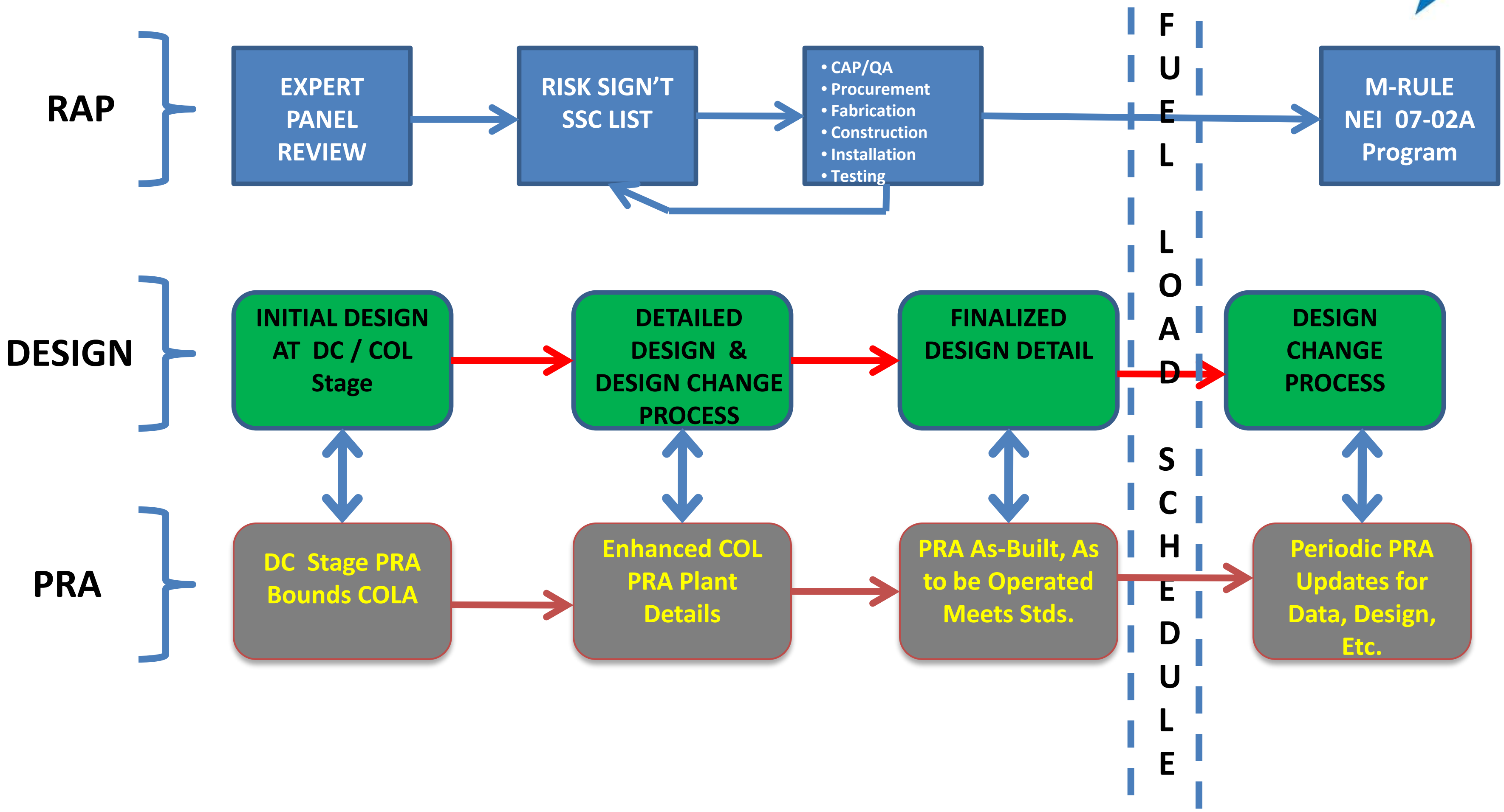
Reliability Assurance
COL Information Items

Item# 17.4-2 continued -- PRA/Design/Expert Panel Interface



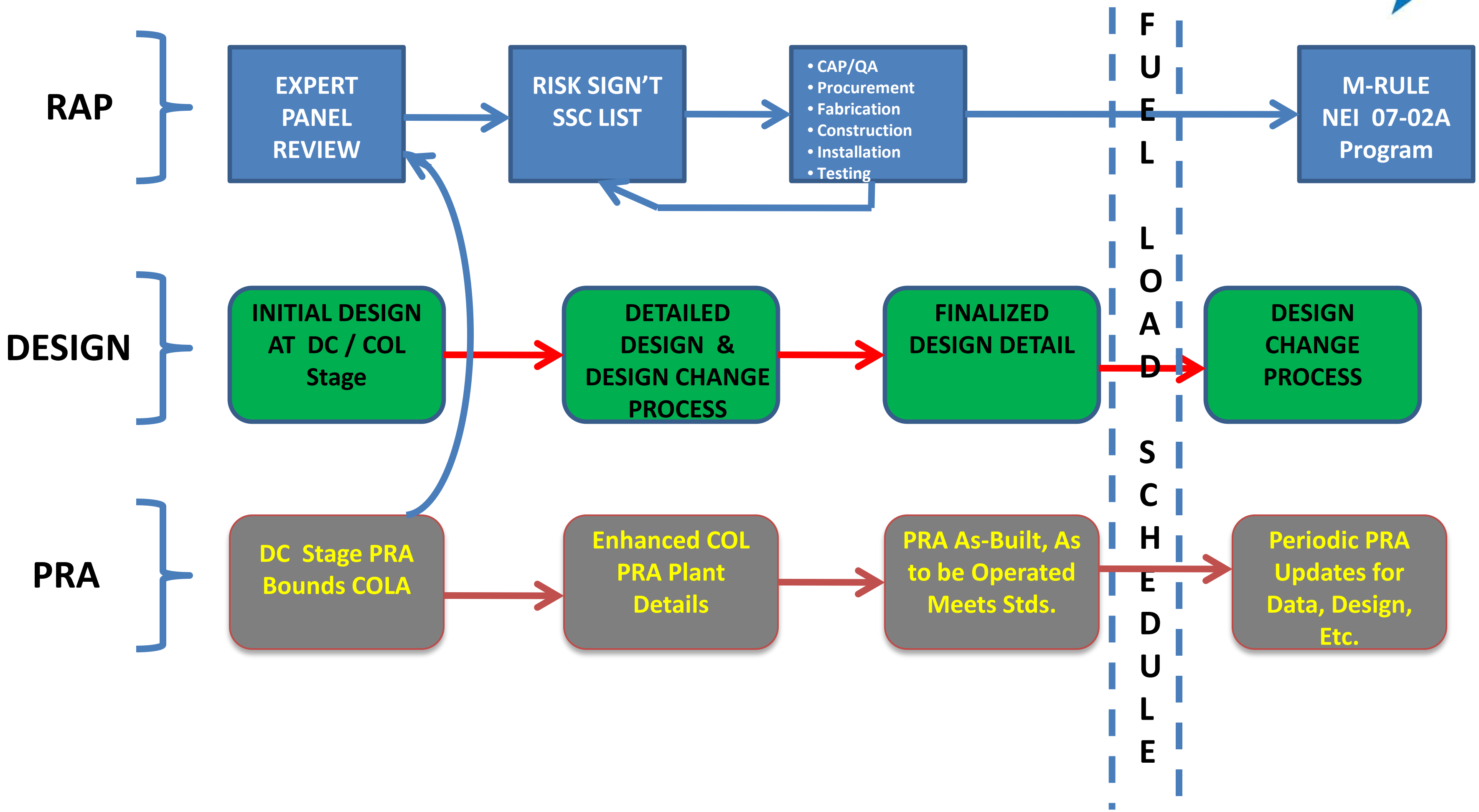
Reliability Assurance
COL Information Items

Item# 17.4-2 continued -- PRA / Design Interface



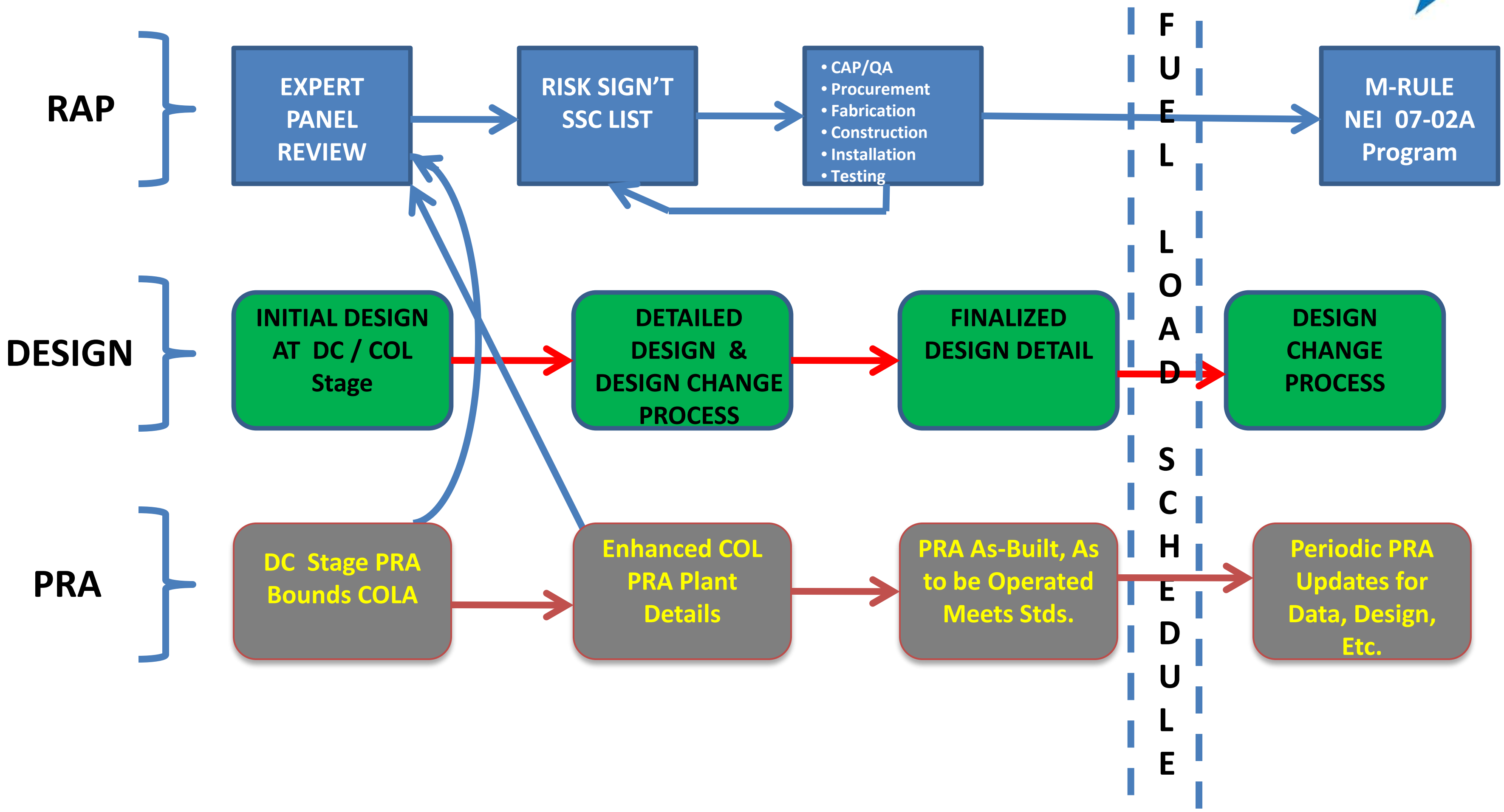
Reliability Assurance
COL Information Items

Item# 17.4-2 continued -- PRA/Expert Panel Interface



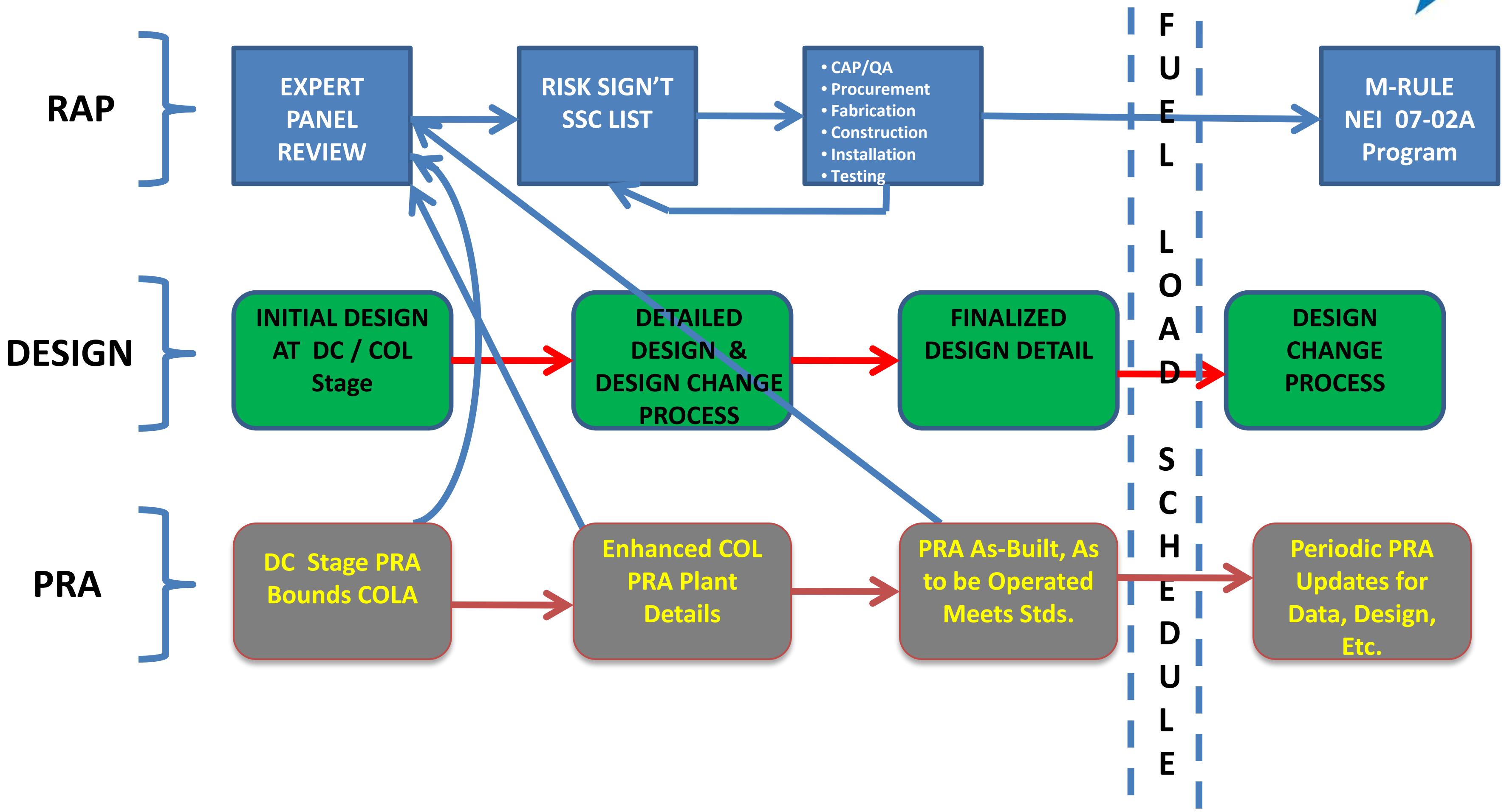
Reliability Assurance
COL Information Items

Item# 17.4-2 continued -- PRA/Expert Panel Interface



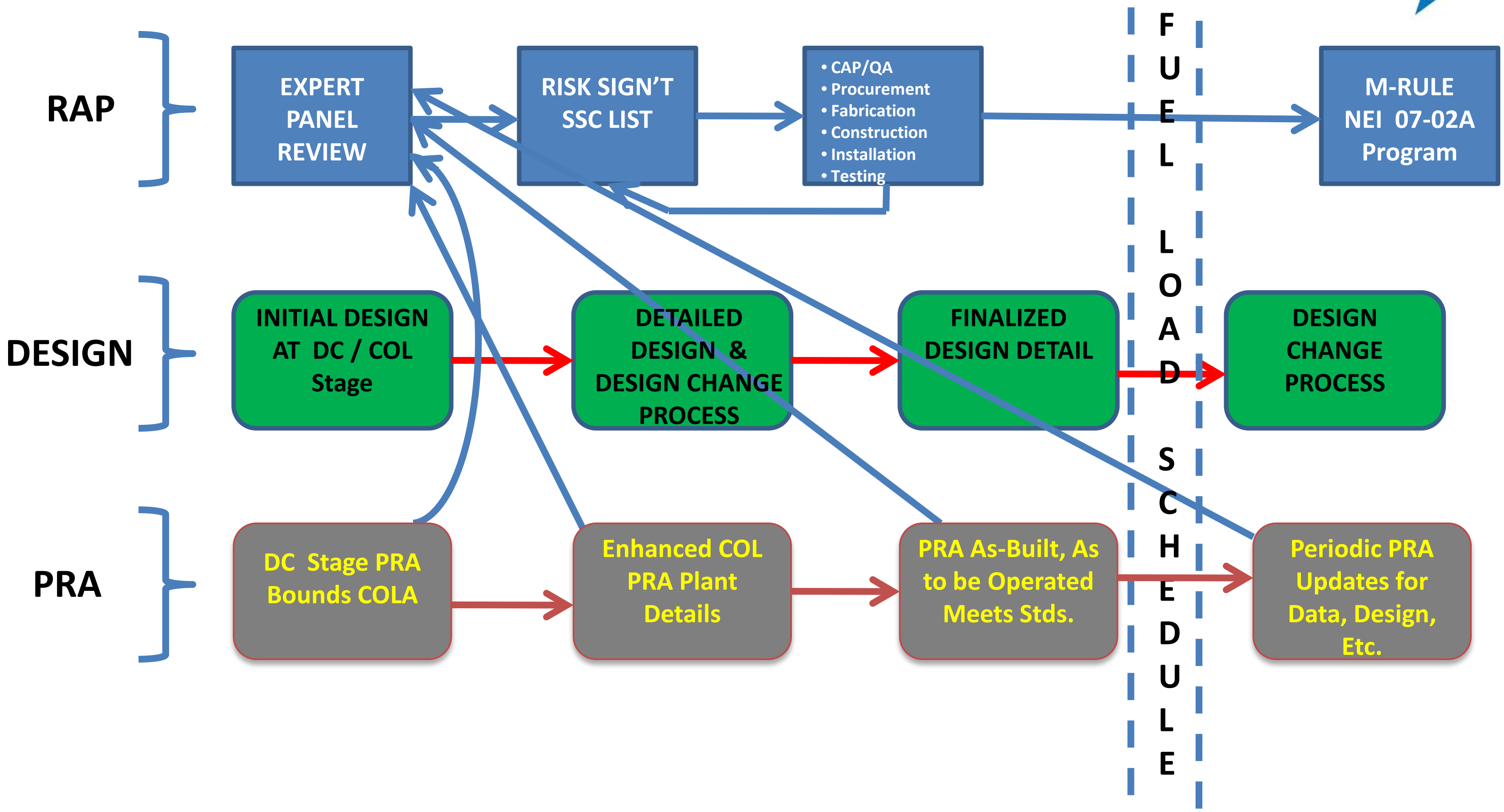
Reliability Assurance
COL Information Items

Item# 17.4-2 continued -- PRA/Expert Panel Interface



Reliability Assurance
COL Information Items

Item# 17.4-2 continued -- PRA/Design Process/Expert Panel Interface



Reliability Assurance
COL Information Items

Item# 17.4-2 continued -- Expert Panel

- RAP Expert Panel – Membership
  - Designated individuals having expertise in the areas of:
    - Risk Assessment
    - Operations
    - Maintenance
    - Engineering
    - Quality Assurance
    - Licensing

- RAP Expert Panel – Categorization
  - Identify a risk categorization of the component based on PRA insights (where the component is modeled)
  - Develop a risk categorization of the component based on deterministic insights
  - Designate the overall categorization of the component

Reliability Assurance
COL Information Items

Item# 17.4-2 continued -- Expert Panel

- RAP Expert Panel – Risk Ranking
  - PRA Ranking: based upon its Fussell-Vesely (FV) importance and its risk achievement worth (RAW)
  - Deterministic Ranking regardless of whether they are also subject to the PRA risk categorization process
    - Is the function used to mitigate accidents or transients?
    - Is the function specifically called out in the Emergency Operating Procedures (EOPs)?
    - Does the loss of the function directly fail another risk-significant system?
    - Is the loss of the function safety significant for shutdown or mode changes?
    - Does the loss of the function, in and of itself, directly cause an initiating event?

Reliability Assurance
NRC SER Open Items

- <u>NRC SER Open Items</u>
  1. RAI 194, Question 17.04-3 (reliability and availability assumptions translated into verifiable attributes), responded to by letter UN#10-029, dated February 12, 2010.
  2. RAI 194, Question 17.04-4 (systems included within the RAP), responded to by letter UN#10-001, dated January 4, 2010.
  3. RAI 224, Question 17.04-5 (describe the system boundary of systems in the RAP), UniStar response transmitted by letter UN#10-094, dated April 16, 2010.
  4. RAI 224, Question 17.04-6 (criteria for selecting the Expert Panel), UniStar response transmitted by letter UN#10-094, dated April 16, 2010.
  5. RAI 224, Question 17.04-7 (rationale for the deterministic categorization process), UniStar response transmitted by letter UN#10-094, dated April 16, 2010.

Reliability Assurance
NRC SER Open Items

- NRC SER Open Items continued

6. RAI 224, Question 17.04-8 (SSCs identified by the deterministic methods (e.g., not modeled in the PRA)), describe the performance criteria and goals), UniStar response transmitted by letter UN#10-094, dated April 16, 2010.
7. RAI 224, Question 17.04-9 (justify the exclusion of certain systems from the scope of D-RAP), UniStar response transmitted by letter UN#10-094, dated April 16, 2010.

Quality Assurance and Reliability Assurance Agenda

- Quality Assurance
  - COL Information Item
  - Oversight Activities
  - NRC SER Open Item
  - NRC SER Confirmatory Item
- Reliability Assurance
  - COL Information Items
  - NRC SER Open Items
- Maintenance Rule
  - COL Information Items
  - NRC SER Open Items
  - NRC SER Confirmatory Item
- Conclusions

Chapter 17, Quality Assurance and Reliability Assurance
Subsections: 17.6, 17.7
Maintenance Rule

Presented by Richard Szoch
UniStar Director of Testing & Programs Development

Maintenance Rule
COL Information Items

- 17.6, 17.7 Maintenance Rule

  - Nuclear Energy Institute Report No. NEI 07-02A, "Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed Under 10 CFR Part 52," Revision 0, dated March 2008, provides the Maintenance Rule Program for Calvert Cliffs Unit 3.

  - NEI 07-02A is incorporated by reference.

Maintenance Rule
COL Information Items

- Item# 17.6-1

  Describe the process for determining which plant structures, systems, and components (SSC) will be included in the scope of the Maintenance Rule Program in accordance with 10 CFR 50.65(b).

- Item# 17.6-2

  Provide the process for determining which SSC within the scope of the Maintenance Rule Program will be tracked to demonstrate effective control of their performance or condition in accordance with paragraph 50.65(a)(2).

- Item# 17.6-3

  Provide a program description for monitoring SSC in accordance with 10 CFR 50.65(a)(1).

- Item# 17.6-4

  Identify and describe the program for periodic evaluation of the Maintenance Rule Program in accordance with 10 CFR 50.65(a)(3).

Maintenance Rule
COL Information Items

Items# 17.6-1 thru -9 continued

- Item# 17.6-5

  Describe the program for maintenance risk assessment and management in accordance with 10 CFR 50.65(a)(4).
- Item# 17.6-6

  Describe the program for selection, training, and qualification of personnel with Maintenance-Rule-related responsibilities consistent with the provisions of Section 13.2 as applicable.
- Item# 17.6-7

  Describe the relationship and interface between Maintenance Rule Program and the Reliability Assurance Program.

Maintenance Rule
COL Information Items

Items# 17.6-1 thru -9 continued

- Item# 17.6-8

  Describe the plan or process for implementing the Maintenance Rule Program as described in the COL application, which includes establishing program elements through sequence and milestones and monitoring or tracking the performance and/or condition of SSC as they become operational.

- Item# 17.6-9

  Describe the program for Maintenance Rule implementation.

These nine COL Items are addressed as follows:

- The plan or process for implementing the Maintenance Rule Program is described in Maintenance Rule Program Implementation description included in NEI 07-02A, "Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed Under 10 CFR Part 52," Revision 0, dated March 2008, which is incorporated by reference for Calvert Cliffs Unit 3.

Maintenance Rule
NRC SER Open Items

- NRC SER Open Items
  1. RAI 192, Question 17.06-2 (Maintenance Rule Program relationship with industry operating experience (IOE) activities), responded to by letter UN#09-485, dated December 4, 2009.
  2. RAI 228, Question 17.06-3 draft received March 29, 2010 (Applicant should propose language for a license condition for the implementation and readiness milestones for the Maintenance Rule program), no response required.

Maintenance Rule
NRC SER Confirmatory Item

- NRC SER Confirmatory Item

  1. Incorporate UniStar response to RAI 62, Question 17.06-1 (justify exclusion of IOE in Section 17.6 of the applicant's FSAR) into COL FSAR 17.6.

  The NRC SER Confirmatory Item is addressed as follows:

  - ➢ UniStar Nuclear Energy will incorporate RAI response into Revision 7 of the COLA.

Quality Assurance and Reliability Assurance Agenda

- Quality Assurance
  - COL Information Items
  - Oversight Activities
  - NRC SER Open Items
  - NRC SER Confirmatory items
- Reliability Assurance
  - COL Information Items
  - NRC SER Open Items
- Maintenance Rule
  - COL Information Items
  - NRC SER Open Items
  - NRC SER Confirmatory Item
- Conclusions

Quality Assurance and Reliability Assurance Conclusions

- COL Information Items, as specified by U.S. EPR FSAR, are addressed in Calvert Cliffs Unit 3 FSAR Chapter 17
- No Departures from EPR FSAR for Chapter 17
- No ASLB Contentions
- Ten NRC SER Open items
- Two NRC Confirmatory Items (Incorporation of RAI response into next COLA revision)

Acronyms

- ACRS-Advisory Committee on Reactor Safeguards
- ACWS-Auxiliary Cooling Water System
- ANSI-American National Standards Institue
- ASLB-Atomic Safety & Licensing Board
- ASME- American Society For Mechanical Engineers
- CDF-Core Damage Frequency
- CFR-Code of Federal Regulations
- COL- Combined License
- COLA-COL Application
- DC-Design Certification
- D-RAP- Design reliability assurance program
- EDF- Électricité de France
- EOP-Emergency Operating Procedures
- EPR- Evolutionary Power Reactor
- FSAR- Final Safety Analysis Report
- FSER – Final Safety Evaluation Report
- IBR-Incorporate by Reference
- IOE -Industry Operating Experience
- JSW-Japan Steel Works
- LRF-Large Release Frequency
- M-Rule-Maintenance Rule
- NEI - Nuclear Energy Institute
- NQA – Nuclear Quality Assurance
- QA – Quality Assurance
- QAPD-quality assurance program description
- Q&PI- Quality & Performance Improvement
- PRA-probabilistic risk analysis
- RAP- Reliability Assurance Program
- RCOLA-Reference COL Application
- SER -safety evaluation report
- SSC-structures, systems, and components
- TR- Topical Report

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

Presentation to the ACRS Subcommittee

UniStar Calvert Cliffs Nuclear Power Plant (CCNPP) Unit 3 COL Application Review

Safety Evaluation Report

CHAPTER 17: Quality Assurance and Reliability Assurance

April 20, 2010

Staff Review Team

- Technical Staff
  - Hanh Phan, Senior Reliability & Risk Engineer PRA and Severe Accidents Branch
  - Jonathan Luciano-Ortega, Operations Engineer
- Project Managers
  - Surinder Arora
  - Tarun Roy

Overview of Staff's Review

SRP Section/Application Section		Number of RAI Questions	Number of SE Open Questions
17.1	Quality Assurance During Design	0	0
17.2	Quality Assurance During the Operations Phase	0	0
17.3	Quality Assurance Program Description	0	0
17.4	Reliability Assurance Program	9	7
17.5	Quality Assurance Program Description	3	1
17.6	Description of Applicant's Program for Implementation of 10 CFR 50.65, the Maintenance Rule	3	2
17.7	Maintenance Rule Program	0	0
Totals		**15**	**10**

Description of Open Item

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Protecting People and the Environment

Section 17.05

- The staff issued RAI 200, Question 17.5-6, to request that the applicant commit to following the guidance in RG 1.33 "Quality Assurance Requirements (Operations) and issue revision1 to UN-TR-06-001-A accordingly.

Description of Open Items (contd.)

Section 17.04

- **RAI 194, Question 17.04-3:** Describe how the reliability and availability assumptions are translated into verifiable attributes as stated in COL FSAR Section 17.4.4.2.
- **RAI 194, Question 17.04-4:** Provide explanation for the exclusion of fire water distribution system, sprinkler system, spray deluge system, and core melt stabilization system from COL FSAR Table 17-4-2 "Design Certification Scope Systems Included within RAP."
- **RAI 224, Question 17.04-5:** Describe the system boundary of the risk-significant systems identified in the COL FSAR Table 17.4-2, "Design Certification Scope Systems Included within RAP," and COL FSAR Table 17.4-3, "Site Specific Systems Included within RAP."
- **RAI 224, Question 17.04-6:** Provide the rationale for the criteria used for selecting the expert panel.

Description of Open Items (contd.)

- **RAI 224, Question 17.04-7:** Provide the rationale for the deterministic categorization process, especially, the classification of weighted score range of 0-40 as a low safety or no risk significance.
- **RAI 224, Question 17.04-8:** Describe the performance criteria and goals for the risk-significant SSCs identified by the deterministic categorization methods.
- **RAI 224, Question 17.04-9:** Justify the exclusion of the normal heat sink, startup and shutdown system, auxiliary cooling water system, closed cooling water system, and raw water supply system from the scope of the RAP.

Description of Open Items (contd.)

Section 17.06

- **RAI 192 Question 17.06-2:** Revise section numbers and COL information item numbers presented in the COL FSAR, Section 17.6 to conform to the most recent revision to the U.S. EPR FSAR.
- **RAI 228 Question 17.06-3:** Provide, as discussed in SECY 05-0197, the implementation and readiness milestones for Maintenance Rule program.

Acronyms:

ASME- American Society for Mechanical Engineers

CCNPP- Calvert Cliffs Nuclear Power Plant

COL - Combined License

NEI - Nuclear Energy Institute

QA - Quality Assurance

QAPD - Quality Assurance Program Description

PRA - Probabilistic Risk Assessment

RAI - Request for Additional Information

RAP - Reliability Assurance Program

RG - Regulatory Guide

SE - Safety Evaluation

SSC - Structures, Systems, and Components

TR - Topical Report

Why Certify New Nuclear Plant Maintenance Programs?

"Nuclear Maintenance Certification: License Plants with a Plan"

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EPR Subcommittee
Advisory Committee on Reactor Safeguards
Rockville, MD
April 20, 2010

Summary

Ref: "Nuclear Maintenance Certification: License Plants with a Plan" (a white paper), by J.K. August & J.J. Hunter, March 2010

I. Why RAP?
II. Part 52
III. What's the intent of Part 52?
IV. What is effective RAP?
V. Benefits
VI. Why now is the time?
VII. What we should do
VIII. Conclusion

After forty years, nuclear plants need scheduled maintenance plans!

Why Reliability Assurance Programs (RAP)?

- Part 50 Experience
  - Ineffective maintenance & unreliable equipment (SSC)
  - Legacy problems, delays and uncertainty
  - Designs sought to assure reliability even before Three Mile Island (TMI). After TMI, they sought it with diligence
  - NRC Operating Experience (Generic Communications)
    - Approximately 300 IE Information Notices, Generic Letters, and Bulletins (IEB) addressed design basis reliability concerns
  - Deterministic designs did not assure reliability

Part 52

- Requires a RAP
- Reasons:
  - Assure PRA reliability
  - Required for Operations
  - Assure safety design basis for certification
- Should answer:
  - What provides an effective RAP?
  - What meets RAP intent, effectively?
  - Design RAP (D-RAP)? Operational RAP (O-RAP)?

Doesn’t answer these completely, today

Part 52 (continued)

- Requirement for RAP is reasonable
- Designers should provide guidance for safety-related SSC
- Development responsibility rests with designers
- Continues the intent of Part 52
- Meeting Part 52's intent would leave less to chance
- Real question: what is a RAP?

What is RAP intent?

RAP should

- Assure nuclear systems, structures and components (SSC) operate with minimum unavailability and the fewest “maintenance-preventable function failures” (MPFF) as required by the Maintenance Rule, 50.65.
- Clearly, it should assure nuclear plant SSC meet design-assumed availability and reliability
- Provide actionable guidance to those who operate & maintain plants

RAP Gap

Difference Present Requirement vs. Effective Program

- Gap Analysis Consequence
- Question is effectiveness and adequacy

What would be effective?

- To be effective, RAP would
  - Address the certified design and the COL
  - Provide tasks that actually make SSC reliable
  - Give clear guidance
  - Complete the Design (e.g., D-RAP)
    - Not just provide lists of equipment
- Specify activities
  - Define specific tasks that make SSC reliable
  - Task/activities must be
    - Clear
    - Explicit
    - *Actionable*
    - Measureable

What would be effective?

- Equipment scheduled maintenance plans
  - completely specified
  - according to standards
  - as part of the certified design
- How?
  - With an effective, consensus-based SSC scheduled maintenance plan development process for complex safety designs. Should provide
    - An effective scheduled maintenance program that becomes the RAP, with
    - Complete, actionable guidance that will assure performance
    - Proven over time, tested with qualified systems and participants
- Nuclear plant programs could specify a much more effective RAP with scheduled maintenance

MSG-3 (2004)

- A documented, well-proven method for developing reliability programs in complex safety applications for over forty years
- Basic development requires:
  1. Identifying critical equipment, critical characteristics and causes of degradation
  2. Developing efficient, effective control tasks on that basis;
  3. Organizing the resulting structured work (composed of actionable tasks) scheduled to implement. [see ADA 066579/DOC-NTIS]
  4. Performing required outcomes – condition directed maintenance

Benefits

- Reduce risk
  - Could substantially reduce nuclear risk
  - Could prepare for new plant staffing
- Standardized reliability programs would
  - meet Part 52 safety intent
  - be more consistent
  - benefit everyone
- Clear RAP would
  - Make projections more certain
  - Lower nuclear costs
  - Improve design and operations consistency

Benefits (continued)

- Complete RAP programs
- Measureable, performable activities
- Fulfill D-RAP, completely
- Current program Inputs (Maintenance Rule etc) Complete
- Leading indicators provided (in contrast with lagging)

Why act now?

- Historical approach confusing & incomplete
  - The licensee was the developer – not the vender (17.4), who had the expertise
  - Finally answer, "What specific, performance-based activities make SSC reliable?"
- Fully address both parts of RAP – design and operations (D-RAP/O-RAP)
- Clarify responsibilities
  - Designers vs. owners
  - Certified design vs. site specific COL
  - Scopes
- Get the best possible certified designs
- Fundamentally support PRA projected
  - Deliver design reliability
  - Less overall cost risk

Why act now?

- Long overdue
  - Common sense
  - Consistent
  - Opportunity now
- Should encourage better methods
  - Safety
  - Design
- Broad benefits
- "Why not?"
  - Why shouldn't we improve RAP, just as we have safety design? Or the QAP? Or Technical Specifications? …..
- Standard nuclear designs (and processes) should improve

How to address RAP Gap

- Perform safety analysis
- Document & share results
  - Safety evaluation reports (SERs)
  - Commission Policy (SECY)
  - Generic Communications (GC)
- According to results, review and revise policy
- Encourage excellence, regardless

What should the ACRS do?

- Take a position
- Recommend action
- Ask for a response from the staff
- Discuss SECYs (referenced) and their basis
- Share responses with stakeholders
- Share conclusions with the Commissioners
- Be transparent!

NRC; Industry?

Alternatives

- Ignore
- Trivialize
- Claim impossibility (“Technology not available” etc.)
- Eliminate requirement
- Preempt

“We’ve always done it,”

“We’d been planning to do it, anyway,”

“We were planning to get to it, when the timing was right, it just hadn’t come yet…” etc, etc

Challenge today

- Improve safety processes
- The safety issue – we're never good enough
- Remove blocking to allow design options
- New plants present a one-time opportunity to get this right
- It will pay for itself; technically feasible – and very simple before construction
- Part 52 needs effective RAP
- We solved this problem forty years (40) ago; why not use it?

After forty years of similar aircraft experience, certifying and using an effective RAP process is just the right thing to do!

References

1. Nuclear Maintenance Certification: License Plants with a Plan" (a white paper), by J.K. August & J.J. Hunter, March 2010
2. Standard Review Plan (NUREG-0800), Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition (formerly issued as NUREG-75/087)
3. RegGuide 1.206, Combined License Applications for Nuclear Power Plants (LWR Edition), Jun 2007 (formerly DG-1145)
4. SECY-93-087, "Policy Technical and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs" fails to meet the Commission's objective of providing adequate guidance to maintain SSC supporting the certified design's PRA.
5. SECY-95-132, "Policy and Technical Issues associated with the Regulatory Treatment of Non-safety Systems (RTNSS)" fails to identify essential D-RAP elements, their development processes or adequacy to support O-RAP operating requirements.
6. SECY 95-132 fails in its intended purpose, too
7. ATA MSG-3(2004), "Operator/Manufacturer Scheduled Maintenance Development", Air Transport Association of America standard MSG-3
8. Approximately eight separate communications with NRC leadership, 2007-2010
9. IAEA-TECDOC-1264, "Reliability Assurance Program Guidebook for Advanced Light Water Reactors", December 2001
10. NUREG-0737, Clarification of TMI Action Plan Requirements (1980)
11. Part 50-Domestic Licensing of Production and Utilization Facilities
12. Part 52-Licenses, Certifications , and Approvals for Nuclear Power Plants