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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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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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APR1400 SUBCOMMITTEE

+ + + + +

FRIDAY

FEBRUARY 24, 2017

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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., Ronald Ballinger, Chairman, presiding.

COMMITTEE MEMBERS:

RONALD BALLINGER, Chair

MATTHEW W. SUNSERI, Co-Chair

CHARLES H. BROWN, JR., Member

MARGARET CHU, Member

WALTER L. KIRCHNER, Member

JOSE A. MARCH-LEUBA, Member

DANA A. POWERS, Member

JOY REMPE, Member

GORDON R. SKILLMAN, Member

JOHN W. STETKAR, Member

DESIGNATED FEDERAL OFFICIAL:

DEREK WIDMAYER

NRC STAFF PRESENT:

LAWRENCE BURKHART, NRO

JEFF CIOCCO, NRO

EDWARD STUTZCAGE, NRO

GETACHEW TESFAYE, NRO

ALSO PRESENT:

SANGHO KANG, KEPCO

JOONKON KIM, KEPCO

TIM LLOYD, WEC

ANDY OH, KHNP

ROB SISK, WEC

IRVING TSANG, DERADS

C-O-N-T-E-N-T-S

P R O C E E D I N G S

8:31 a.m.

CHAIR BALLINGER: (presiding) Good morning. The meeting will now come to order.

This is a meeting of the APR1400 Subcommittee of the Advisory Committee on Reactor Safeguards.

I'm Ronald Ballinger, Chairman of the -- and I've never been called that before -- I'm Ron Ballinger, Chairman of the APR1400 Subcommittee.

ACRS members in attendance today are Joy Rempe, Charles Brown, Jose March-Leuba, John Stetkar, Matt Sunseri, Dana Powers, Gordon Skillman, Margaret Chu, and Walt Kirchner.

Derek Widmayer is the Designated Federal Officer for this meeting.

The purpose of today's meeting is for the Subcommittee to receive briefings from Korea Electric Power Corporation and Korea Hydro and Nuclear Power Company, Limited, regarding their Design Certification Application and the NRC staff regarding their Safety Evaluation Report with Open Items specific to Chapter 12, Radiation Protection.

ACRS was established by statute and is

governed by the Federal Advisory Committee Act, FACA. That means that the Committee can only speak through its published letter reports. We hold meetings to gather information to support our deliberations. Interested parties who wish to provide comments can contact our offices requesting time after the meeting announcement is published in The Federal Register. That said, we also set aside 10 minutes for spur-of-the-moment comments from members of the public attending or listening to our meetings. Written comments are also welcome.

The ACRS section of the U.S. NRC public website provides our charter, bylaws, letter reports, and full transcripts of all full and subcommittee meetings, including slides presented at the meetings.

The rules for participation in today's meeting were announced in The Federal Register on Tuesday, February 7th, 2017. The meeting was announced as an open/closed-to-the-public meeting. This meant that the Chairman can close the meeting as needed to protect information propriety to KHNP or its visitors. I understand that there's no proprietary information.

No request for making a statement to

the Subcommittee has been received from the public.

A transcript of the meeting is being kept and will be made available, as stated in The Federal Register notice. Therefore, we request that participants in this meeting use the microphones -- better than I do, actually; press the little button until the green light comes on -- located throughout the meeting room when addressing the Subcommittee. Participants should first identify themselves and speak with sufficient clarity and volume, so that they be readily heard.

We have a bridgeline established for interested members of the public to listen-in. The bridge number and password were published in the agenda posted on the NRC public website. To minimize disturbance, this public line will be kept in a listen-in mode only. The public will have an opportunity to make a statement or provide comments at a designated time towards the end of this meeting.

I request that the meeting attendees and participants silence cell phones and other electronic devices as well.

I invite now Jeff Ciocco, NRO Project Manager, to introduce the presenters and start the briefing.

Jeff?

MR. CIOCCO: Yes. Thank you.

My name is Jeff Ciocco. I'm the Lead Project Manager for the APR1400 Standard Design Certification Application.

Thank you for having us back today for the sixth APR1400 Subcommittee meeting on Chapter 12, Radiation Protection. We're here with our technical staff and our Project Managers, and we are ready to go.

Thank you.

MR. SISK: This is Rob Sisk, Westinghouse, and on behalf of KHNP and KEPCO, we appreciate this opportunity to present Chapter 12.

Without more, I will turn it over to Mr. Sangho Kang, and we'll go from there.

MR. KANG: Thank you.

Good morning, everyone.

My name is Sangho Kang. I was working as a nuclear engineer and group supervisor at KEPCO Engineering and Construction.

Today I am going to talk about radiation protection and the features of APR1400. I am very pleased to have this opportunity to present the overview of the DCD Chapter 12 to ACRS members.

Before I start my presentation, I would like to introduce my team. On my left there is Mr. Dongsu Lee who is the Team Lead of Radiation Protection at KEPCO E&C. On my right, sitting, Mr. Irving Tsang from DERADS, is our technical consultant for Chapters 11 and 12. Mr. Joonkon Kim, sitting next to Mr. Lee, is the INC Team Lead at KEPCO E&C, is going to make presentation for the Area Radiation Monitoring System.

Now I can start the presentation. This presentation follows the other sections of DCD Tier 2 Chapter 12, as shown in this slide.

After a brief overview of the application and review status of Chapter 12, I will talk about the highlights of each section from 12.1 through 12.5. And then, I will show you the list of COL items which belong to this chapter. Then, I will talk about the open items and summarize this presentation.

For the staff's review of radiation protection design features, we submitted DCD Tier 1 and Tier 2 without any technical or topical reports. The total number of RAIs associated with Chapter 12 is 83. We have submitted all the responses as of

February 23rd, this year.

According to the staff's SER, there are 14 open items. We are working with the staff to close these open items. So, I'm going to talk about the details of the open items at the end of this presentation.

The structure of Chapter 12 follows the guidance of Reg Guide 1.206 and the associated SRPs.

Section 12.1 discusses about the design and operational policies to ensure that the occupational radiation exposures are maintained ALARA.

Section 12.2 provides the information on the radiation sources in the plant, including the contained, airborne, and accident source terms.

The APR1400 radiation protection design features such as layout, systems design, shielding, ventilation, and area radiation monitors are providing in Section 12.3.

Section 12.4 presents the dose assessment on occupational radiation exposure and the vital area mission doses and the design features to minimize contamination and radwaste generation.

The Operational Radiation Protection Program is described in Section 12.5.

Now I'm going to start with the APR1400 policies to assure ORE are ALARA. In order to ensure that occupational radiation exposures are ALARA, APR1400 provides organizational structure to effectively implement radiation protection policy, training, and reviews that are consistent with operational and maintenance requirements.

There are several Regulatory Guides related to ALARA implementation, including Reg Guide 1.8, 1.33, 8.8, and 8.10.

The design policy of APR1400 is to implement the ALARA philosophy during the early stages of the design. This is fulfilled through the design review and documentation that assure consistency with the APR1400 ALARA design guide. The design is then supplemented by operational policies and programs that are intended to keep occupational exposure ALARA.

Details of the design considerations for maintaining ORE ALARA are shown in the second bullet of this slide. First, the APR1400-specific ALARA Design Guide, which is a high-level design criteria document developed in accordance with the associated Regulatory Guides, specify the approach, methods, and implementation guides for the responsible engineers to take into account when they prepare, check, and review the design documents.

CHAIR BALLINGER: I have a question about that. This title APR1400 ALARA Design Guide appears in a number of places in the chapter, but we don't have that. Is that an available document?

MR. KANG: It was not requested by the staff, but that's the Design Guide we use internally.

CHAIR BALLINGER: Oh, okay. Because it seemed like it was the high-level --

MR. KANG: Yes.

CHAIR BALLINGER: -- document that sort of underwrote everything.

MR. KANG: Right.

CHAIR BALLINGER: Thank you.

MR. KANG: And the lessons learned from the construction and operation of the earlier nuclear power plants are incorporated using the systematic design procedures.

MEMBER SKILLMAN: Would you give us some examples, please, of the lessons learned that have been incorporated in your APR1400, lessons learned from your other nuclear power plants?

MR. KANG: Yes. Yes, because we have

constructed that in the nuclear power plants since the 1970s, we have a lot of comments from the construction workers and operators in terms of the ALARA to reduce the dose. So, these kinds of comments are collected and provided to the designer. Because in Korea we are one family, the operator and the designer, this information was delivered to us, so that we can take into it account for the next construction and design.

CHAIR BALLINGER: So, are these lessons learned, are they incorporated into this ALARA Design Guide? Does that get updated?

MR. KANG: That kind of lessons learned are not controlled by this ALARA Design Guide, but the process is described in this guide. And the details, the systematic system, which is operated by the designer and collecting information from the site, we will check it and we implement it, incorporate their comments into the next design. That's our design procedure.

MEMBER SKILLMAN: Okay. That's very nice. Can you give us an example of where the APR1400 has been changed because of what you learned from your prior experience?

MR. KANG: I don't remember exactly.

But, for example, the access control system --

MEMBER SKILLMAN: The what?

MR. KANG: Access controls system, access control system for checking the entrance to the radiation area, and there is a health physics room which we designed. But, during the operation, the operator might feel that this is not convenient with respect to the exposure control. Then, they give us the comments. Then, we take into account their comments. We change our design by incorporating their comments. So, that's the one example.

MEMBER SKILLMAN: Are there any other examples that are prominent? For instance, the relocation of passageways or the relocation of shielding?

MR. KANG: That could be one of the comments. Yes. I don't remember exactly, but we can provide some kind of list for you after we get back home.

MEMBER SKILLMAN: No, I was just interested in how the design is maturing based on the experience that you have to date. That was the tone of the question. I'm not requesting a list. Do you understand what I'm --

MR. KANG: Yes, the list, the comment lists are piled up in our system, and we check it for the start of the new construction design. Because those kinds of comments were raised by the operator. So, they ask us to incorporate their comments.

MEMBER SKILLMAN: I was particularly interested in whether or not you chose to reroute high-energy piping or piping that might contain a very high source term to a different location based on the experience that you have had in your other plants to date. That was what I was really curious about.

MR. KANG: So, the high-energy line which contains the highest radioactive source as we can -- I don't remember exactly. But, normally, we take the main steam line as a high-energy line; also, the similar blowdown and the others. But those kinds of system components that are part of the secondary system, and some part of the CVCS might be -- I don't know, exactly remember --

MEMBER SKILLMAN: The letdown line.

MR. KANG: Yes.

MEMBER SKILLMAN: The letdown line.

MR. KANG: Yes, it's that kind of piping

should be routed within the pipe chase which is shielded by the big complete doors.

MEMBER SKILLMAN: Thank you.

MR. SISK: This is Rob Sisk.

Just additional information for you. You may want to take a look at Section 12.1.3.2, which provides design features for the ALARA during maintenance and inspection. And it talks about some of the features that have been incorporated into APR1400.

MEMBER SKILLMAN: Thank you, Rob.

MR. KANG: Okay. The ALARA training program, which is a part of the ALARA Design Guide, is also an effective way of implementing ALARA during the design process.

In the design of equipment, the ALARA Design Guide requires to select the materials to effectively remove the contamination; to enhance reliability; to reduce maintenance, and to minimize corrosion.

In the layout design, the guide also requires to separate the radioactive equipment from the non-radioactive equipment and to provide sufficient area for inspection and maintenance.

The details of the ALARA design

considerations are described in Section 12.3 of the DCD, as Rob just mentioned.

The operational ALARA considerations are not within the scope of the Design Certification and will be provided by the COL applicants.

There are no outstanding review items by the staff for this section.

If you do not have any questions, I can move forward, move onto the 12.2. 12.1 is about the radiation sources of the APR1400.

In this slide, I will talk about the sources in the nuclear steam supply system first, the primary radiation emanating from the reactor core and also from the reactor vessel during normal operation and neutrons and gamma rays produced by the fission reactions.

The reactor core fission products are estimated using ORIGIN-S computer code based on the thermal power of 102 percent. The fission product core inventory data is provided in Chapter 15, Appendix A, not in Chapter 12.

The source of radiation in the reactor coolant system are the fission products released from the fuel and the activation and corrosion products. The fission product source terms provided in Chapter 12 are determined based on the fuel defect rate of .25 percent and used for the design of shielding and ventilation.

MEMBER POWERS: Does a calculation based on the defect give us a useful understanding of the contamination in the reactor coolant system with respect to activation and corrosion products? I can understand using it for shielding purposes, but if I want to understand what the activity is within the reactor coolant system, don't I need some other mechanism to understand activation and corrosion products, especially since I think at our last meeting we discussed extensively the use of cobalt alloys in the system?

MR. KANG: The core products in the reactor coolant system are produced by the activation of cobalt and other impurities within the reactor coolant system material. If it is corroded and you have some particles which go through the reactor core and it is activated and becomes radioactive, that is the mechanism of the production of corrosion products.

In the design of the APR1400, the amount of corrosion product in the RCS system is based on the operating experience in the United States, and the amount of corrosion product in the system was used as our shielding design as well as for the HVAC and the affluent design.

MEMBER POWERS: Okay. You have succeeded in puzzling me enormously.

(Laughter.)

MR. KANG: Yes, we understand that it is quite complicated to analytically calculate it. And we know that there are several models which can calculate the corrosion product inside the RCS, but we do not rely on the analytical model for the use of the design basis source. Instead, we use operating experience data.

CHAIR BALLINGER: I am sure this is going to come up more during this discussion, but if you will look at the DCD and some of the discussion in the SER for the Open Items with respect to dose, .25-percent failed fuel, which is a number you can use for shielding calculation, that is 50 fuel rods failed. Under normal operation, there is no way that any utility would ever operate with that many failed fuel rods, not even more than two, right? They would shut down and get rid of it.

And yet, in one section of the DCD it says that 25 to 50 percent of the dose is due to failed fuel, and the other 50 percent, or it adds up to 100 or so, is due to corrosion products. So, if that is true, and you actually have operating data on operating plants for which I'm sure you don't have 150 failed fuel rods, how does that square with the shielding calculations, the measurements that you actually do in a plant when you say this is my calculated dose for design purposes, but this is what I actually measure?

Maybe I'm not stating it as clearly as I could. But I'm wondering about that because it almost seems like if, in fact, 25 to 50 percent of the dose is from failed fuel and the failed fuel fraction is an artificially-high number, then when you actually operate the plant, 50 percent of the dose is due to something that will never happen. And so, if you actually measure the dose, it should be much, much lower. Is that true?

MR. KANG: I can tell you --

CHAIR BALLINGER: I'm sure I'm maybe not being clear.

MR. KANG: Yes. Yes, you're right, we use .25-percent fuel defect for shielding design basis because it is required by the SRP and Reg Guide 8.8.

CHAIR BALLINGER: Yes.

MR. KANG: If you want to use the lower number, then we have to justify. But we don't want to tackle with the regulation. So, that is why we use .25 percent as --

CHAIR BALLINGER: Yes, I understand that part, but when you actually go and measure the dose in room X --

MR. KANG: Yes.

CHAIR BALLINGER: -- for which you have calculated a number --

MR. KANG: Yes, when we actually look at the operating experience there for occupational exposure, then most of the dose comes from the corrosion product for the activity --

CHAIR BALLINGER: Right.

MEMBER KIRCHNER: Yes, that makes sense, yes.

CHAIR BALLINGER: Yes, it makes sense, yes.

MEMBER KIRCHNER: That makes sense.

MEMBER SKILLMAN: What is the span for normal operation of fuel? It must be substantially below .25 percent.

CHAIR BALLINGER: Well, they will shut

down with one failed fuel, two failed fuel rods. They will put them out, right?

MEMBER SKILLMAN: Right. It is really all governed by dose-equivalent iodine in your tech specs. And DEI will tell you what you need to do. In just a skosh of a pinhole leak or a failed fuel pin you will be there already. It will drive you to the dose-equivalent iodine, DEI, and you will shut down.

MEMBER KIRCHNER: So, going back to Dick's question earlier, from your operating experience, did you change any materials to reduce corrosion or as a source term in your APR1400 design?

MR. KANG: Because the cobalt is the main source of the corrosion products, in our design specification we limit the contents of cobalt in the RCS components to less than 7 amount of level. That is how we control the production of the corrosion products. And the vendor, it is describing the CDS --

CHAIR BALLINGER: Yes, my understanding, that that was subject to one of the RAIs, which we might be discussing later on, where certain materials, the cobalt content was actually reduced as a result of an RAI.

MEMBER POWERS: But, still, if memory serves, we have got a lot of cobalt in this.

CHAIR BALLINGER: There are cobalt alloys used in selective locations.

MEMBER POWERS: For hardening.

CHAIR BALLINGER: For hard-facing and stuff.

MEMBER POWERS: And we end up not having a good understanding of what the real contamination in this system is. I mean, it seems to me using .25-percent fuel defect for shielding calculations is just fine, but to understand what the real contamination in the system is, we have got to have something different than that.

CHAIR BALLINGER: Yes. And for the record, I went back and looked at the AP1000 system. After all the RAIs and things like that --

MEMBER STETKAR: Be careful.

CHAIR BALLINGER: Okay. Never mind. I was never here.

(Laughter.)

MEMBER STETKAR: Just be careful about --

MEMBER POWERS: It's a different system.

It is a different plant, different power.

MEMBER STETKAR: And it might be proprietary.

CHAIR BALLINGER: Okay.

MEMBER POWERS: But they are consistent. Material traces are consistent.

MEMBER KIRCHNER: So, then, therefore, what is the assumption, like the second bullet on fuel defect, what is the assumption for generating the corrosion source?

MR. KANG: As I mentioned, the corrosion product, the amount of corrosion products used in the design are based on the operating experience. So, we do not make any assumption to calculate the corrosion product source term.

MEMBER KIRCHNER: You haven't operated long enough to get what the amount would be after 40 or 60 years, right?

MR. KANG: No, because the --

MEMBER KIRCHNER: So, you must, then, extrapolate some estimate of what the corrosion products are?

MR. SISK: This is Rob Sisk, Westinghouse.

And perhaps maybe I can at least

understand the question more. The APR1400 is in operation in Korea. It is based on the OPR1000, which has a tremendous amount of operating history.

The level of cobalt is comparable between the two units. So, what they have been able to do is go to actual operating history, actual operating measurements of the plants in Korea and make assumptions based on comparable levels of cobalt. It is the same fuel, +7 fuel. And based on that, they are able to bring that knowledge or that information into their assessments for the APR1400. So, there is practical operating experience on which this is based.

MEMBER REMPE: Just a second, though. I thought earlier I heard you say or one of the members here say that it was based on U.S. operating experience. Is it U.S. operating experience or Korean +7 fuel operating experience?

MR. KANG: It is based on U.S. experience in the operating plants in the 1970s, because that makes our design more facilitative because the test source and corrosion source were to be used for the design basis of surety and ventilation. So, in actual measurement data in the Korean operating plant the sourcing would be much lower there.

MEMBER REMPE: So, you used U.S. experience, but you have benchmarked those numbers with Korean operating experience and found that it is very conservative and your Korean plants are running with lower amounts, to paraphrase?

MR. KANG: Yes.

MEMBER REMPE: Thank you.

MR. SISK: The designs are very similar across the board, as we have talked about with CE and OPR.

MEMBER REMPE: Yes.

MR. SISK: So, recognizing U.S. operating experience, which is different than Korea -- I think you have summarized it very well.

CHAIR BALLINGER: I should probably know this, but what is the steam generator tubing material in the OPR1000?

MR. KANG: OPR1000? That is the Inconel 600.

CHAIR BALLINGER: Okay. So, it is alloy 600?

MR. KANG: The APR1400 is 690.

CHAIR BALLINGER: Yes, well, there's a big difference between OPR1000 and APR1400. And that will have a big effect on corrosion product transport and concentration and stuff. So, that is a huge difference between the two.

MR. KANG: Yes, but I don't remember exactly what material was used for OPR1000 for the steam generator, but if they use Inconel 690, it is one of the design improvements --

CHAIR BALLINGER: Right.

MR. KANG: -- to reduce the dose for APR1400.

CHAIR BALLINGER: That's good.

MEMBER KIRCHNER: So, that is a good answer to Dick's earlier question where you have made a substantive change to reduce dose in the spirit of ALARA, right?

MEMBER SKILLMAN: I think the design changes --

MEMBER KIRCHNER: Or whatever you were asking, things like that --

MEMBER SKILLMAN: It's not just operating dose rates. It is also the very real need to change out steam generators.

MEMBER KIRCHNER: Exactly.

(Laughter.)

MEMBER SKILLMAN: That's real ALARA

savings. There's where your ALARA savings is. So, if you don't have to change them out, then you are lightyears ahead in terms of the worker dose.

CHAIR BALLINGER: But I think the pressure boundary is about 75 or 80 percent in the steam generator, right?

MEMBER SKILLMAN: Yes, of the heat transfer area.

CHAIR BALLINGER: Yes.

MEMBER MARCH-LEUBA: Just out of curiosity, and you may not know the answer to this, but I assume the instrumentation in the plant has a gross dose rate measurement in which you set your alarms if you surpass that dose rate. But, at least periodically, you do some galvanized spectroscopy. So, you do the gamma spectrum function of energy. So, you know where that dose is coming from, which you can separate iodine from cobalt from nitrogen-16. Is that correct?

MR. KANG: During normal operation, in the design we do not have that kind of gamma spectroscopy permanently in the plant. We can use it as a portable equipment. So, if it is necessary --

MEMBER MARCH-LEUBA: And I'm sure you

do. If you get an alarm, you will go on, say, high scramble or a high --

MR. KANG: Yes. We have personal dosimetry.

MEMBER MARCH-LEUBA: Sure.

MR. KANG: Yes, and, also, area radiation monitors, which is how much everywhere.

MEMBER MARCH-LEUBA: But my question is, if you went to .25-percent failed fuel versus --

MR. KANG: Well, that is controlled by the technical specification, limited condition of operation.

MEMBER MARCH-LEUBA: Yes.

MR. KANG: It's the operator takes a sample from the RCS, and it is sent to the laboratory.

MEMBER MARCH-LEUBA: Oh, you do like a --

MR. KANG: They check --

MEMBER MARCH-LEUBA: It's an instructive sample --

MR. KANG: Right.

MEMBER MARCH-LEUBA: -- of the water?

MR. KANG: Yes. They check it, whether it exceeds the RCL limit or not.

MEMBER MARCH-LEUBA: Are there problems with coatings on the pipes? I mean, some of those corrosion products will transport and they will not show up in a sample of the water.

MR. KANG: As I know, we do not require --

MEMBER MARCH-LEUBA: No?

MR. KANG: -- to have electric polishing of the surface.

MEMBER MARCH-LEUBA: Okay. Thank you.

CHAIR BALLINGER: But that would be an option.

MR. KANG: Yes.

CHAIR BALLINGER: We will probably never get beyond slide 8 if we keep going. But there was a question there of electric polishing versus fine machining. And in most cases, you did not decide to do electric polishing?

MR. KANG: Not yet.

CHAIR BALLINGER: Not yet?

MEMBER REMPE: Before you leave this slide, could we discuss the DAMSAM -- if I am pronouncing it correctly -- code? It is not an approved code. I believe in some areas the NRC said, well, we'll just compare it with hand calculations and other values we had. But, eventually, you're comparing it with the Westinghouse codes which are approved.

MR. KANG: Yes, and that is what I am going to explain in the key issue items in a couple of slides later.

MEMBER REMPE: Okay. Could you explain a little bit about what the code does? Or will you talk about that later? Because how it determines the species of the isotopes released and things like that --

MR. KANG: I can turn that question to the Westinghouse team.

MEMBER REMPE: Okay, and whenever you want to do it is fine, but I just would like to hear more about the code.

CHAIR BALLINGER: Tell us your name, please.

MR. LLOYD: My name is Tim Lloyd. I'm with Westinghouse.

I can either begin to answer that now -- I think there's a slide coming up in one or two that would be a good place to dovetail.

MEMBER REMPE: That's fine. I just want to make sure I get to understand.

MR. LLOYD: So, we will. We will come to that.

MEMBER REMPE: Okay. Thanks.

MR. KANG: Can I continue?

So, I talk about the source term in Chapter 12. And another design basis source term determined based on 1-percent fuel defect is provided in Chapter 11. This source term is used for the system design, equipment qualification, and accident analysis. The RCS source term is calculated by DAMSAM code.

One of the water activation products, nitrogen-16, is the predominant radiation source in the reactor coolant system due to it high-energy gamma. However, since its half-life is very short and the design provides sufficient time for decay inside the containment, it is not a significant source outside the containment.

The spent fuel assemblies are the predominant source of radiation in the reactor containment building after plant shutdown for refueling. The most significant sources in the auxiliary building, except for the spent fuels, are contained in the CVCS components. The design basis source terms are determined assuming that the CVCS gas stripper is not operated to maximize the gaseous source terms. The CVCS source term calculation is performed using SHIELD-APR computer program. The source terms for the Shutdown Cooling System are calculated assuming that the system starts operation after four hours after shutdown.

Now I am going to talk about the auxiliary system. The design basis source terms in the secondary systems, including main steam, steam generator blowdown, and condensate polishing systems, are determined assuming that the primary coolant is leaking by .6 gallons per minute in the two steam generators. This assumption is considered conservative since the limiting condition of operation or LCO for steam generator leakage rate is .2 gpm.

And the source terms in the Component Cooling Water System is determined assuming that all of the unidentified RCS leakage of .5 gpm for an hour is transferred to the CCWS system. The unidentified leakage of .5 gpm for an hour is also the LCO defined in the technical specifications.

The design basis source term in the spent fuel pool water is determined assuming that the primary coolant water is mixed with the spent fuel pool water after 48 hours of shutdown cooling operation. And the maximum source terms in the filters, ion exchangers, and the associated piping of the system are calculated using this initial spent fuel source term.

The source terms in the Liquid Waste Management System are determined based on the expected in-flow rates and their activities into the three kinds of LWMS collection tanks. The buildup activities in the treatment components are then calculated using DIJESTER computer code. In particular, the source term for the monitor tank, which is used to collect the treated water and sampling for final discharge, is determined based on the maximum level of primary coolant.

The buildup activities in the Gaseous Waste Management System charcoal delay beds are calculated using the inflow from CVCS tanks and the gas stripper.

The solid wastes, such as spent filters and resins and the reversis osmosis sludge, are generated in the CVCS, LWMS, Spent fuel Pool Cooling and Cleanup System, and Steam Generator Blowdown Systems. Source terms in resin storage tanks in the SWMS are calculated based on their in-flow source terms. The source terms for the dry active wastes are calculated based on the maximum operating experience data and the storage plan.

Now I am going to talk about the airborne and accident source terms. The source of airborne contamination mostly comes from the leakage from the radioactive systems, including CVCS, radwaste, and HVAC systems. Other sources are evaporation of pool water and some vents from the tanks.

The airborne activity calculations are performed in all radioactive areas to determine the minimum required HVAC flow rates to maintain the airborne concentration ALARA. The ventilation flow rates in areas which require frequent access determined to maintain the airborne concentrations are less than a small fraction of derived air concentration, or DAC, specified in 10 CFR Part 20, Appendix B. Other areas are designed to less than 1 DAC.

The accident source terms are defined to provide adequate shielding in vital areas during accident conditions, so that the operators are protected during their post-accident mitigation actions. In addition, the accident source terms are used to determine the environmental conditions for the equipment important to safety.

As recommended in Reg Guide 1.183, the alternative source term which assumes a significant core melt condition is used to define the accident source term. Shielding for the areas which require continuous occupancy such as the main control room and the technical support center is designed to maintain the average dose rate over 30 days is less than .15 millisieverts per hour as well as meeting the cumulative dose of 50 millisieverts, as required in GDC 19.

MEMBER POWERS: I guess I don't understand, certainly don't understand the line concerning areas requiring infrequent access. I just don't know what you mean exactly there.

And I would also like, for continuous occupation, what is the annual accumulated dose of an operator in the main control room if you are operating at your limit?

MR. KANG: During the accident condition, right? During the accident conditions? You are talking about the MCR dose during the accident.

MEMBER POWERS: I see.

MR. KANG: Yes, that is described in DCD Section 6.4, which is about MCR habitability. Also, this is provided in Chapter 15 in LOCA Section 15.6.5. Because the requirement is 15 millisieverts for the whole period of the accident. It is described there. I don't remember the exact number, but it is 40-somewhat millisieverts for 30 days.

CHAIR BALLINGER: I recall that there was an RAI related to this. It came out to be like 49.78 or something like that --

MR. KANG: Yes, that is for the --

CHAIR BALLINGER: -- which was suspiciously close to 50.

MR. KANG: Yes, that is not for the MCR.

CHAIR BALLINGER: Okay.

MR. KANG: That is for the other infrequent access area.

CHAIR BALLINGER: Okay. Thanks.

MR. KANG: Did I answer your question?

MEMBER POWERS: I understand better now. Thank you.

MEMBER REMPE: Before you leave that slide, on the partitioning of the nuclides and activity concentrations, I believe you and the staff have had some exchanges and you eventually went back to NUREG-0409 to justify how much of the halogens were -- how much was the super-halogens in the cold liquid and hot liquid. Do you have any data to support that assumption, too, or you solely relied on this old, very old NUREG?

MR. KANG: Because we would like to use the data which are approved by the U.S. NRC, even though we might have some kind of information, we did not want to use our experimental data, something like this. So, that is why we used the NUREG data and the EPRI information. That is how we responded.

MEMBER REMPE: Okay. I will ask the staff if they have confidence in that very old NUREG then, when they get up. Okay? Thank you.

MR. KANG: Yes. Thank you for your question.

Let me continue. For other vital areas which do not require continuous occupancy, the shielding design aims at meeting 50 millisieverts during the time for taking post-accident actions.

The key review item for the Section 12.2 source term is about the consideration of daughter nuclides. The staff requested to provide justification why the source terms are already more conservative than they would be if the contribution of daughters was included.

In order to estimate the effects on the buildup of the daughter products in the RCS and the CVCS, we reviewed the methodology of DAMSAM and the SHIELD-APR code system and compared with the results of another code system which takes into account the daughter nuclides. The review demonstrated that there are sufficient conservatisms in the results of the DAMSAM and SHIELD-APR code system.

For the other systems, including spent fuel pool cooling and cleanup system, blowdown system, polishing system, and the gaseous waste system, the source terms considering daughter nuclides were conservatively evaluated and the shielding analyses were performed again using the updated source terms. As a result, the impacts of the new source terms on the current design were negligible since the civil structure design has sufficient margin to bound the minor increase of the source terms. That's how we responded in the RAI and the staff is under review of our responses.

MR. LLOYD: So, Sangho, I think this is the right time to jump in and answer Dr. Rempe's question.

MR. KANG: Yes.

MR. LLOYD: Yes, and for starters, you had asked essentially what DAMSAM does that kind of sets us up to be able to look at it on a comparison basis with the Westinghouse equivalent codes, right?

MEMBER REMPE: Okay.

MR. LLOYD: I have a slide, when I discuss this stuff, that kind of shows what -- by the way, this is Tim Lloyd again with Westinghouse -- they go from. In our analysis for the Westinghouse plants we would look at ORIGEN and, then, go to FIPCO and, then, a code called SSP. ORIGEN I think most people here are aware what it does. It usually is used to track burnup in fuel and to produce all the radionuclides that exist in the fuel.

We, then, model that in a way that leaks out into the reactor coolant system and also gets processed into a few components, volume control tank and gas decay tank. And then, we move into a code called SSP which looks at a whole bunch of different auxiliary components.

What happens in APR1400 is that they use a code called DAMSAM which effectively does the job of ORIGEN and also that initial leak part out into the reactor coolant. And then, they use SHIELD or

SHIELD-APR for that plant to do all the other complicated -- looking at every different component, every different tank, and mainly in the application it gets used for the CVS or CVCS system. So, that's background.

MEMBER REMPE: Okay.

MR. LLOYD: If I can speak to why we think that DAMSAM does not have a problem with non-conservatism, there are two reasons. One reason is that there is no difficulty with daughter treatment in DAMSAM. It treats daughters properly and completely. There's no difficulty there. And we do demonstrate that.

The best way to get a demonstration is to look at reactor coolant system activities. As a proxy for that, we were able to find between our typical plants and the APR1400, "our" being Westinghouse plants, and the APR1400 plant, there were four really good comparisons between these codes and into tanks.

And so, one example, it is almost a trivial example, except it is a great benchmark, is the letdown heat exchanger. And that is just effectively full of the reactor coolant system. We looked at that with a total of four different ways of modeling, four different types of cases. And it does always bump up. There is no difference.

The conclusion, for a number of reasons, was that there isn't difficulty under the hood with DAMSAM. It's pretty straightforwardly doing its job. There is also some methodological things that they do where they run for five cycles to make sure they build up enough inventory that, when they then pull the maximum everywhere, you get maximum values. The difficulties where there were problems with the treatment of daughters were all in the SHIELD-APR code.

MEMBER REMPE: Okay. Thank you.

MR. LLOYD: Okay.

MR. KANG: So, then, I will move on to 12.3. And we will talk about the radiation protection design features addressed in Section 12.3.

This section covers the ALARA design features, shielding and ventilation design, and area radiation monitoring system.

Mr. Joonkon Kim will present the area radiation monitoring system, and I will be back to cover the remaining items.

The APR1400 design incorporates the

ALARA principles in accordance with Reg Guide 8.8 and 8.10. The principles are applied to various areas of the plant design including: layout, equipment and system design, source term control, airborne control, radiation zoning and shielding, and the vital area accessibility.

Details of the design features to ensure ORE ALARA are provided in DCD Section 12.3.1.

The design criteria for shielding are specified in Reg Guide 8.8, 40 CFR 190, GDC 19, and 10 CFR 50.34.

In order to ensure compliance with these criteria, the shielding analyses are performed using several computer codes such as ANISN, MCNP, and MicroShield, which are widely used in the design of nuclear facilities. The RUNT-G code is used to determine the post-accident shielding requirements.

The shielding analyses produce the design basis drawings, including radiation zone maps and the minimum required shield thicknesses. Details of the design information is provided in DCD Section 12.3.

As mentioned in the previous slide, the ventilation flows are provided to ensure the airborne contamination is maintained less than DAC fractions. These airflows are designed to flow from lower to higher contaminated areas, so that the spread of contamination is minimized.

In addition, the continuous air monitoring by effluents and area radiation monitors ensure the detection of airborne contamination level change within 10 DAC-hours, as required in the SRP 12.3.

If you don't have other questions, I would like to turn the microphone over to Mr. Kim for the Area Radiation Monitoring System.

MR. KIM: Good morning, ladies and gentlemen.

My name is Joonkon Kim of KEPCO E&C.

My presentation will discuss the Area Radiation Monitoring System design features and description of system functions. During my presentation you may ask questions at anytime.

The purpose for ARMS design is to warn operators and station personnel of unusual radiological events to protect personnel from radiation exposure in radioactivity or contaminated areas. The ARMS monitors normal radiation levels as well as post-accident radiation levels in selected areas in the plant.

The ARMS meets the applicable requirements of 10 CFR 20, 10 CFR 50, 10 CFR 70, NUREG-0737, Reg Guide 1.97, and ANSI/ANS-HPSSC-6.8.1.

Location of ARMS monitors is determined based on expected frequency of crew access, occupancy time, and potential radiation levels in the plant work areas. The monitors are installed where access to safety-related equipment is required during post-accident conditions. The ARMS provides visible and audible alarms and readouts in the main control room and the local areas when the radioactivity level exceeds a predetermined setpoint. Portable radiation monitors are used for the plant personnel to determine airborne iodine concentration and the optimal route to vital areas to minimize personnel exposure as low as reasonably achievable.

Containment upper and lower operating area monitors and spent fuel pool area monitors are safety-related monitors in accordance with the safety criteria of ANSI/ANS-51.1. Other monitors are non-safety-related.

Two redundant safety-related containment upper operating area monitors detect high-range gamma radiation after a design basis accident to meet the requirements of Reg Guide 1.97.

Two redundant safety-related containment lower operating area monitors monitor fuel-handing accidents.

These four containment area monitors initiate a containment purge isolation function when a high radiation level is detected. This isolation prevents the radioactive materials to be released outside containment.

When a fuel-handing accident occurs, two redundant safety-related spent fuel pool area monitors initiate the fuel-handing area emergency ventilation upon detection of a high radiation level in the spent fuel pool area.

Warning and alarm setpoints of the ARMS will be determined by the COL applicant after the site-specific conditions and operational requirements are determined.

The ARMS monitor consists of a detector part, which is RE, for detecting gamma radiation and electronic or display part, which is RT, for processing and displaying of the radiation signl. The RE consists of ionization chambers or Geiger-Mueller tubes. The RE and RT are separated or combined on a single skid, depending upon the monitoring location.

All local alarms are located at the electronic part of the monitor. Some monitors have a local alarm at the detector part as well as the electronic part, depending upon the monitoring locations.

Okay, next slide.

APR1400 design has 16 places where a total of 21 ARMS monitors are installed, as shown in this table. Each monitor has its safety function, electrical class, seismic category, measuring range, function, and display/alarm location.

Containment upper operating area, containment of lower operating area, and spent fuel pool area monitors are designated as safety-related monitors. They are designed to comply with redundancy requirements to meet the single failure criteria of IEEE Standard 603.

The containment upper and lower operating area monitors, in-core instrument area, hot machine shop area, truck bay area, and the waste drum storage area monitors have local alarms both at the detector part and the electronic part of the monitor.

AMI stands for accident monitoring instrumentations which is required by Reg Guide 1.97. The operator can assess the plant status following design basis events by monitoring plant variables and equipment/system operating status.

The containment upper operating area, containment lower operating area, and spent fuel pool area monitors are designed as AMI Type C. Other AMI monitors are Type E, in accordance with Reg Guide 1.97 and IEEE Standard 497.

This is my presentation of ARMS. Thank you for your time.

MR. KANG: Okay. I am Sangho Kang again. So, I will continue with the key review items for Section 12.3

The key review item for this section is RAI 8599, Question 12.03-53. The question requests to provide the cumulative impacts on overall radiation protection design by the change of normal operation source terms due to consideration of daughter nuclides and the change in accident source term.

As mentioned in a previous slide, we provided the response to Question 12.02-22, 23, and 25, addressing that the consideration of daughter nuclides has no impact on the current physical design.

Since the accident source term was modified during the RAI process, the vital area mission dose analyses were updated. And it was found that the mission doses for all vital areas meet the criteria of 50 millisieverts.

The response to this RAI was just submitted and it is under the staff's review now.

MEMBER MARCH-LEUBA: Sorry, I must have been asleep before when you were presenting this. Can you explain what the dose limit of 15 millisieverts is? Is that the dose to the whole population in the plant or per person, per day, per year, per life of the plant?

MR. KANG: Yes, this is for the accident condition as defined in the 10 CFR 50, Appendix A, where the operator could take emergency action.

MEMBER MARCH-LEUBA: To a single operator?

MR. KANG: Yes.

MEMBER MARCH-LEUBA: Okay. So, that's to a single operator for an event?

MR. KANG: Yes.

MEMBER MARCH-LEUBA: And that's 5 rem,

for those of us who think American.

(Laughter.)

MR. KANG: The dose to the operators, the workers in the United States as defined in ICRP-20, which is 15 millisieverts.

MEMBER MARCH-LEUBA: Yes, that is the personal dose maximum allowed, yes. We typically stay much lower than that.

MR. KANG: Yes. Thank you for your question.

Okay, I'm going to move on to the next slide.

The Section 12.4 addresses the dose assessment for occupational exposure and vital area mission dose. And it also provides design features to minimize contamination and radwaste generation.

The regulatory guidance for ORE estimation is specified in Reg Guide 8.19. Since the first APR1400 has just started its commercial operation in Korea, the ORE estimation for APR1400 was based on the measurement data for an operating plant, Hanul Unit No. 3, which is also the combustion engineering type PWR. And this exposure data was increased taking into account the power ratio.

Even though the APR1400 has several dose-reduction design improvements described in DCD Section 12.4.1.1.1, this was not taken into account in the ORE estimation. The resultant ORE was estimated to be 585 person-millisieverts per year or 58.5 person-rem per year.

MEMBER MARCH-LEUBA: Okay. I'm following up on my question. Can you explain those units? Is this the dose for the whole population of the plant?

MR. KANG: The terminology we use is collective dose. That is for the many people.

MEMBER MARCH-LEUBA: Right. And roughly, you are talking 5 people, 50, or 500?

MR. KANG: It's a couple of hundred people.

MEMBER MARCH-LEUBA: A couple of hundred? Okay. Thank you.

MR. KANG: As specified in Reg Guide 8.19, the DCD table provides the details of the breakdown of the work tables, so that it can provide the dose-causing activities to facilitate further reduction efforts for construction and operation.

CHAIR BALLINGER: I have a question. Back on the routine maintenance fraction, it is about a quarter of the dose. We had an earlier comment about the difference between electric polishing and fine machining. In the U.S. there's a push for electric polishing to reduce crud deposition and things like that. For your design, you are going to avoid that. You are going to use fine machining. Did you do a detailed analysis of the pros and cons and how much you might save in terms of dose for electric polishing versus using what we call fine machining, I guess? I mean, how was that decision made?

MR. KANG: I might not be the right person to answer this question because it is strongly related to the operation. I'm from KEPCO E&C, the designer, and KHNP, who is the operating plants, might have the answer for your question. So, we can table that question and we will get back to you later.

CHAIR BALLINGER: Yes. I'm just curious, is it a 10-percent savings or would it be a 50-percent savings? Because that is a lot of dose.

MR. KANG: Yes. Well, we designers have not performed those kind of analyses yet.

CHAIR BALLINGER: Do any of the OPR1000 plants or Hanul have -- is there a comparison in

Korea between machining and electric polishing that you can get information?

MR. KANG: I don't have that information, no.

MR. SISK: This is Rob Sisk.

We have the wrong people here in the room to answer that question at this time, Ron.

CHAIR BALLINGER: Okay.

MR. SISK: We could take a look to see what we have available, but we are not prepared to address that today.

MEMBER MARCH-LEUBA: Off the top of my head, if you have 200 people involved and roughly 15 rem per year, you are having 250 millirem per year for the average person. It means that a few of them are getting 500 or more, and that's high. It is not negligible.

MR. KANG: The 585 millisieverts is the total dose.

MEMBER MARCH-LEUBA: Correct, but you said, roughly, 200 people were involved.

MR. KANG: Yes.

MEMBER MARCH-LEUBA: So, that means you get a quarter of a rem per year per person, for the average person, meaning that some of them are going to get twice as much.

MR. KANG: The situation might be similar in the United States. The individual dose is below 10 millisieverts per year. So, the 3 to 5, that is the highly probable individual dose to the people who are working in these --

MEMBER MARCH-LEUBA: If I remember my training correctly, whenever you hit 500 millirem in a year, you are not allowed to work anymore. You get sent to a desk, right? Is that correct?

MEMBER POWERS: Yes, typically.

MEMBER MARCH-LEUBA: Because, I mean, that's 10 percent of the maximum dose, which is 5 rem.

MEMBER POWERS: Five rem, yes.

MEMBER MARCH-LEUBA: So, I mean, anything you can do to reduce that -- I assume that this is a high-bound estimate? Does this correspond to actual operating conditions?

MR. KANG: The number came from the actual operating data, and we increased it by a power ratio of 1.4 because the data came from the 1,000-megawatt PWR. But we didn't take into account the design enhancement which was incorporated in the APR1400 design, not to mention the use of Inconel

690, and the additional design features. We can say that we have an integrated reactor head assembly to facilitate the removal we would have during the refueling operation.

MEMBER MARCH-LEUBA: Oh.

MR. KANG: They can save time for the exposure, something like that. But, if you take into account that kind of design enhancement, we can reduce this number. The actual number would be lower than this one.

MEMBER MARCH-LEUBA: Yes, I would expect it to be much lower.

MEMBER SKILLMAN: I would like to explore this a little further. In the past number of years in this country, the very clean Pressurized Water Reactors generate about 5 to 10 rem per year. A lot of plants are at 5 rem per year, 5,000 millirem, for the entire site population.

MR. KANG: That's man-rem per year, right?

MEMBER SKILLMAN: That's correct.

MR. KANG: Yes.

MEMBER SKILLMAN: Person-rem.

MR. KANG: Yes, because if you do not have any refueling outage, the dose would be much lower.

MEMBER SKILLMAN: Exactly, exactly. So, hold that thought, and that's exactly where I was going. So, you subtract the 163 millisieverts per year from your 585, that is about 400 millisieverts a year. That's very different than the experience in this country for the very clean Pressurized Water Reactors. It's almost a factor of 100.

So, what's the difference? Is this your actual operating data from Korea?

MR. KANG: Yes, it comes from the Hanul operation data from 2004 to 2013. And the refueling, in this calculation we have taken into account the overhaul for refueling outages and it was factored in, in order to give the one-year exposure. Because the APR1400 has the fuel cycle of 18 months. So, this number is for the 12 months.

So, all the contributing, dose-contributing activities are included here, based on the operating experience. And I can mention you might have lower collective dose in a U.S. operating plant, but, based on my knowledge, the KHNP who is operating our plant has very good numbers, low-dose annual exposure by a good performance. So, they are trying their best to reduce the dose.

But I didn't exactly compare it this number with the U.S. operating numbers yet. So, we can go back and check what the real actual status in the United States and what is the status of our calculation compared to your operating experience.

MEMBER SKILLMAN: I think the reference that you will use is NUREG-0713. This is the most recent, and that is what is driving me to ask this question.

I understand your 585 sieverts are 58.5 person-rem. What I'm observing is that for the last number of years the very clean Pressurized Water Reactors, particularly those with the Inconel 690 tubes --

MR. KANG: Yes.

MEMBER SKILLMAN: -- no source term, the radiological controls people are controlling at the millirem level on a daily basis. And at the end of a calendar year, 12 months, many of the plants are below 10 person-rem, 10,000 MR. And so, that number is starkly different from what would be approximately 400 person-rem on a non-outage year.

MR. KANG: Okay. It might be dependent on what the power of the plant is. And then, the information from that literature might come from some lower power then.

MEMBER SKILLMAN: I'm talking 1,000-megawatt electrical plants.

MR. KANG: A thousand?

MEMBER SKILLMAN: Four-loopers, three-loopers, large plants.

MR. KANG: Because we increased by multiplying 1.4 from the 1,000-megawatt data. So, if we want to compare with your data, then we have to divide by 1.4 again to compare with your numbers. And then, if they have incorporated some kind of zinc injection, you might have a much lower dose. But this number didn't take into account zinc injection. As I know, the zinc injection can reduce the dose by 20 to 30 percent. And the number could be similar.

MEMBER SKILLMAN: Okay. Thank you.

MR. KANG: So, it might be different.

MEMBER SKILLMAN: Okay.

CO-CHAIR SUNSERI: I want --

CHAIR BALLINGER: Go ahead.

CO-CHAIR SUNSERI: One of the practices that I think in the United States is leading to these low numbers -- well, actually, never mind. I'm thinking about the outage dose reduction, not the online operation.

But let me finish my question anyway. So, I did not see -- one of the operating practices in the United States for outages is to perform what is called a crud-burst cleanup. So, a crud-burst is induced at the beginning of the outage. The corrosion products are removed and the radiation doses are significantly reduced during the outage.

I did not see where that practice is invoked in your plants. Is that a strategy or not?

MR. KANG: Unfortunately, I cannot answer that question.

So, can you answer, Irving?

MR. TSANG: My name is Irving Tsang.

You have a description of the different activities performed to come up with dose in the chapter. We could go back and look at that and see that activity, whether this crud-burst cleanup activity is specifically included or not. Off the top of my head, I do not remember.

MR. KANG: But the data we got from the operator does not include what condition at that time.

MR. SISK: This is Rob Sisk, Westinghouse.

I would make one observation here that maybe is helpful. A lot of what we are talking about is operational parameters of how the plant is operated and how the plant is maintained by the COL. The ALARA program and the operational considerations really fall into a COL area where they will make decisions on whether to use any condition, crud-burst, things like this that they use to enhance their plant activities.

So, I guess I just wanted to bring that up to kind of make sure we focus on what the design requirements are which are one level and, then, of course, we expect the numbers to be much less as you go forward and you implement a robust ALARA program from a licensee perspective.

MEMBER SKILLMAN: Rob, I appreciate that, but that is why my first question was, what operating experience or what changes have you made into the APR1400? These numbers, if these numbers are truly anticipated, I would expect your shielding design to be enhanced. That's why I asked the question. If you really think you are going to -- I'm going to say it -- burn through that much radiological exposure, I would expect that in certain areas you would be doubling your shielding because you can get these numbers down to a reasonable level. At some point there are diminishing returns on the investment; I understand that.

But in all of the passageways and all of the places where operators need to go, I have been part of dose reduction and it is a discipline. It is how you think about where people are going to be and when, how they get there and get back, and what it takes to get there. Do they have to be in scuba? Do they have to be breathing air packs? Is the ventilation going to prevent exposure? And are the walls thick enough for what is on the other side to be sufficiently shielded for the isotopes?

MEMBER MARCH-LEUBA: Another item to consider is this is person-rem per year.

MEMBER SKILLMAN: Yes.

MEMBER MARCH-LEUBA: So, the U.S. plants are operated with a very small amount of people.

MEMBER SKILLMAN: Well, that's what I said. These plants --

MEMBER MARCH-LEUBA: If they use twice as many workers in there, then we will get twice as much --

MEMBER SKILLMAN: Bingo. A clean P with

690 tubes is running 110 rem on a not-outage year without any failed fuel, squeaky-clean fuel, generators tight. You are running under -- some of them are running 5 rem per year. And the HPs are managing it, 2- and 3- and 4-millirem per week and month.

MEMBER MARCH-LEUBA: The point I was trying to put on the record is a control issue that some people -- you are using two workers to do a job and, then, in the U.S. you would send only one. Right there, you just doubled your dose.

MEMBER SKILLMAN: Or not do the job at all.

MEMBER MARCH-LEUBA: Or don't do it, correct.

So, there are places for savings there that you need to consider.

CHAIR BALLINGER: Since you opened the door -- (laughter) -- and you mentioned zinc, the plant does not have, if I recall reading right, a setup to do the zinc injection, is that correct?

MR. KANG: We don't have zinc injection, no.

CHAIR BALLINGER: Okay. Why?

MR. KANG: I think the COL applicants,

if they want, they can do it.

CHAIR BALLINGER: Okay.

MR. KANG: Applicants can question if it is necessary to reduce the dose.

CHAIR BALLINGER: Okay. Thank you.

MR. KANG: Okay. And I would like to conclude this slide. The ORE was estimated to be 585 person-millisievert per year. As specified in Reg Guide 8.19, the DCD table provides the details of the breakdown of the work activities, so that it can provide the dose-causing activities.

CHAIR BALLINGER: Sorry again. The Hanul Unit 3 is more than 10 years old. So, the chances are that has alloy 600 tubing for steam generators? Or have the steam generators been replaced?

MR. OH: This is Andy Oh, KHNP Washington office.

Hanul No. 3 is, based on my memory, is their steam generator is replaced. At the time, I think their material was also replaced with 690 TT or something like that.

CHAIR BALLINGER: But how long ago?

MR. OH: That's about four or five years ago.

CHAIR BALLINGER: Okay. So, 10 years is probably half the time is with an alloy 600 steam generator?

MR. OH: I guess --

MR. KANG: I think so.

CHAIR BALLINGER: Okay. So, that complicates things just a little bit if you are trying to figure out what might happen with alloy 690 then.

MR. KANG: Right. Yes.

So, let me move on to the vital area mission dose. As mentioned earlier, the vital area mission doses are estimated for both continuously-occupied areas and the infrequent access areas.

The infrequent access areas include: post-accident sampling area; remote shutdown room and remote control console room; Class 1E switchgear room; I&C equipment room; access areas outside the containment spray and shutdown cooling pump rooms.

The exposure to the plant personnel who take emergency action is calculated using the source terms, transit and stay time in the course of the access route to the vital areas. As indicated in DCD Table 12.4-8, the estimated mission doses satisfy the regulatory limit of 50 millisieverts.

MEMBER SKILLMAN: Did you assume for the past post-accident sampling system the quarter-percent failed fuel? Is that your source term for that?

MR. KANG: No. For this calculation, we assume significant core melt, which is defined at 1.183. That is the severe accident condition.

MEMBER SKILLMAN: Yes. You can't get to the pass if you have had big core damage -- I can tell you that for sure -- unless you are wearing lead clothing. I mean, if you have really had an accident, you've got to make sure that the pass is well-shielded.

MR. KANG: Yes. The post-accident shielding is based on that source term because we have the core melt reactor and the IRWST is contaminated by the fission product. And the safety injection pump and containment spray pump is taking the water from the IRWST, which is highly contaminated. But these components are located at the bottom level of the auxiliary building. And the operators should go through these areas, and these areas, these components are heavily shielded by considering the core melt source term --

MEMBER SKILLMAN: Okay. Thank you.

MR. KANG: -- to get into the sampling station.

MEMBER STETKAR: Let me ask you -- I've gotten a little confused about the post-accident dose rates that you show in the figures. And you're going to have to stop me if I get into any proprietary information. So, keep me honest, please.

In particular, when we reviewed Chapter 10 of the DCD, I asked a question about personnel access to the main steam atmospheric dump valves, the MSADVs, which are located in the main steam valve rooms. And I was told at that time that that is not a problem because there is good shielding there and they are fully accessible.

I have not completed my review of the PRA, but I am confident that the PRA takes credit for operators manually/mechanically operating those valves with local handwheels. I was told in Chapter 10 that those handwheels are, in fact, located at the valves, which is not surprising.

If I look at the -- and it's figure 12.3-36 in the DCD -- I see post-accident dose rates in those locations on the order of 100 millisieverts per hour. It is relatively high if I have to stand

and manually operate those valves for an extended period of time to control the cooldown after the accident, which the PRA includes credit for.

So, I am curious now about what types of shielding are available at those valves and whether they really are accessible for the times that are required, as included in the PRA for manual operation of those valves. You don't have to answer that today, but it is one area that I was particularly interested in. And all of your other evaluations of areas requiring infrequent access you say should be less than 50 millisieverts total equivalent dose. I couldn't find any discussion of those particular areas, locations in the plant, except for this figure.

So, if you could take that back and try to reconcile that with what credit may be taken in the PRA, for example, for local manual/mechanical operation of those valves? Because there are scenarios in the PRA that require an extended cooldown for several hours by control of those valves.

MR. SISK: We have captured such, but I'll --

MEMBER STETKAR: Thank you, and I hope I

didn't say anything that was proprietary.

MR. KANG: Okay. So, we captured the question. But, as I look at the figures, the 1 millisievert per hour in that area, in the main steam valve areas --

MEMBER STETKAR: I'm sorry, that's normal operation. I'm looking at figure 12.3-36, which is one hour after the accident. It shows up to 100 millisieverts per hour in those areas.

MR. KANG: I'm looking at the same figure.

MEMBER STETKAR: Are you? Okay.

MR. KANG: Yes. But it is true that in that area after one hour of the accident, it is less than 100 millisieverts power, which is very high.

MEMBER STETKAR: Yes.

MR. KANG: But, if we look at the later time --

MEMBER STETKAR: I don't care about the later time because all of the accidents in the PRA -- all of the requirements are to initiate a 100-degree-Fahrenheit-per-hour manual cooldown of the secondary side, with the idea of achieving low pressure in the primary side. And that has to be initiated within a relatively short period of time.

So, I don't care about things like several days later or a week later. I care about shortly after these events occur.

MR. KANG: My one question is, do they have to operate the valve before core melt or after core melt?

MEMBER STETKAR: In the PRA it is -- I don't know the complete answer to that question. Most of the scenarios that I think I know, because I have not seen the details of the PRA models, I think most of them are pre-core melt. However, there are some depressurization post-core-melt scenarios. I don't know whether they account for the secondary cooldown.

MR. KANG: Yes, that is what I was asking.

MEMBER STETKAR: I don't, I just don't know.

MR. OH: Yes, this is Andy, KHNP Washington office.

That MSADV manual operation is only used for the SCBO status, not core is affected or melted.

MEMBER STETKAR: Pre-core-melt? Okay.

MR. OH: Yes, pre-core-melt.

MEMBER STETKAR: Are these dose rates

post- -- these accident dose rates and these figures are all post-core-damage or are they --

MR. KANG: Right, because we assume --

MEMBER STETKAR: Okay.

MR. KANG: -- a significant core melt.

MEMBER STETKAR: Okay. I didn't appreciate that. So, these dose rates are all --

MR. KANG: Right.

MEMBER STETKAR: -- post significant core melt? Thanks. That answers my question. Thank you. No problem.

MR. KANG: So, can I continue for the minimization of contamination? So, as you are aware of, 10 CFR 20.1406 requires to provide how the plant is designed to minimize contamination of the facility and the environment and to minimize generation of radioactive waste. The detailed guidance to implement 10 CFR 10.1406 are provided in Reg Guide 4.21.

To fulfill these requirements, APR1400 established six high-level design and operational objectives. The first four items under the second bullet are those applied to the design, and the remaining two are applicable during the operational phase.

The four design objectives include: prevent and minimize contamination of the facility and the environment; provisions for the leak detection to support a timely and appropriate response in the event of an unintended release of radioactive contamination; and provision of the capability to reduce cross-contamination; the need for decontamination, and the generation of radioactive waste. And the final one is to provide decommissioning planning.

The two operational objectives are: development of the operations and documentation and development of the site radiological environmental monitoring.

Under these four design objectives, the APR1400 design was reviewed and evaluated in accordance with the design guidance in Reg Guide 4.21. As a result, an extensive list of design features are identified and provided as a table in Section 12.4.

This slide shows an example of early leak detection capability of APR1400. In order to detect any leakage in the LWMS tank room, the floor is sloped to a drainage port, of which pipe flows out of the room. The pipe is equipped with the water detection instrumentation, of which signal is provided to the plant operator. The leakage is, then, routed to the sump inside a trench for further treatment.

And next is 12.5. This Section 12.5 covers the operational RP program.

Next.

This operational RP program is to be developed and implemented by the COL applicant to maintain occupational and public doses meet the regulatory limits and are maintained ALARA. There is no open item in this section.

All of the 19 COL items in Chapter 12 are listed throughout the following three slides, but I am not going to talk about the details of these COL items. There are three COL items in Section 12.1, just one for 12.2, four items in 12.3, 10 items in 12.4, and one item in 12.5.

Now I am going to discuss about the open items. There are 14 open items in the staff's SER with Open Items. Five items belong to Section 12.2 and the rest, nine items, are for Section 12.3. There is no item for Sections 12.1, 12.4, and 12.5.

As shown in this and the next slides, responses for most of the open items have been submitted. For some of the items, the staff is under review of our responses, and for the others, we are working with the staff to resolve some additional comments to close these items. So, you can take a look at the current status of the open items from these slides.

MEMBER SKILLMAN: Sangho, let me ask this. This has to do with the third item there on the outdoor tanks. I was interested in the design of the holdup tank, the boric acid storage tank, and the reactor water makeup tank, particularly the holdup tank. That is what we call, we used to call it a push-pull tank. We could move an awful lot of boric acid out of the reactor coolant system when you were changing the reactor coolant system chemistry.

But there isn't any real information about that tank. There is a sentence in Chapter 9. It is on page 9.3-54 and 65. But there isn't any real description of that tank.

MR. KANG: About how the tank, it looks --

MEMBER SKILLMAN: How it is shielded.

MR. KANG: How the shield --

MEMBER SKILLMAN: Yes. Apparently, it

is outside. It is on a pad. It appears to have a concrete shield exterior to the three tanks, and the three tanks sit together, the holdup tank, the boric acid storage tank, and the reactor water makeup tank. But there is no description.

MR. KANG: Yes, but, actually, that is part of the detailed design. So, we did not provide this shielding information for those yard tanks.

And in our design, the yard tanks are shielded just around the outside exterior of the tank with the concrete, like that way. That is how we shield the yard tanks.

And the cover is not shown there for the main tanks purpose, because these areas are not going to impact the dose to the people because it is high enough not to expose the people.

MEMBER SKILLMAN: I just make that observation. You have described a tank that is important to the operation of the plant, but there isn't any real information. There is the table, but there is not a whole lot of description.

MR. KANG: The shielding for this, the yard tank, is a part of the detailed civil structure design which is not the scope of the design certification.

MEMBER SKILLMAN: Okay. Thank you.

MR. KANG: Thank you for your question.

If you don't have any further questions, I would like to close the presentation. In summary, we can say that the policy and design considerations applied to APR1400 conform to the associated Regulatory Guides.

The radiation sources, which are based on .25-percent fuel defect, are used in the design and provided in Section 12.2, in accordance with the SRP 12.2.

The radiation protection design features to ensure that ORE are ALARA are consistent with the guidance in Reg Guide 8.8.

The ORE doses are estimated based on the operating experience data and is provided in the DCD.

And the vital area mission doses are within the criteria in GDC 19 and NUREG-0737.

Design features to minimize contamination and waste generation comply with the requirements in 10 CFR 20.1406 and Reg Guide 4.21.

We have 14 open items that were identified by the staff's SER. Even though most of the responses have been submitted, there are items that need further review or discussion. We are working with the staff to close these open items.

That's all for my presentation for Chapter 12, and thank you for your attention.

MEMBER REMPE: I had a question about one of your responses, and I just want to understand it. It was a question, I believe, that you asked, Ron, of: do you have zinc injection? And I believe your response was, well, the COL applicant can add that if they want, but it is not part of the certified design. Is that your response? Is it in the OPR1400? In Korea is it part of other designs you have sold and established?

MR. KANG: In the original design for OPR1000 the zinc injection was not there.

MEMBER REMPE: Uh-hum.

MR. KANG: And as I know, the one operating plant has implemented, has modified their design to include the zinc injection. That could be Hanul. I don't remember.

MEMBER REMPE: Okay. Thank you.

MR. KANG: I'm sorry, I don't remember which unit has incorporated zinc injection.

MEMBER REMPE: Okay. Thank you.

CHAIR BALLINGER: Questions, further

questions? Further questions?

(No response.)

No?

We are eight minutes ahead of schedule, which is remarkable. So, we will recess until, let's try 20 minutes after.

(Whereupon, the foregoing matter went off the record at 10:07 a.m. and went back on the record at 10:22 a.m.)

CHAIR BALLINGER: We are back in session. We are back in session. The floor is yours, whoever "you" is.

(Laughter.)

MR. TESFAYE: Okay. Good morning, everyone.

My name is Getachew Tesfaye. I'm the NRC Project Manager for APR1400, Chapter 12, Radiation Protection. Ed Stutzcage, on my right, is the technical reviewer, and, of course, you have heard from Jeff Ciocco earlier, who is the Lead Project Manager.

The staff has completed the first review and submitted the report to you about a month ago. As you have heard earlier in the Applicant's presentation, there were 14 open items that will be addressed in the next phase, Phase 4. Today Ed will go through these four major bullets and give you the areas that he reviewed and his group reviewed, and we will also address their findings, and, finally, the remaining issues.

So, with that, I will leave the slides.

MR. STUTZCAGE: All right. Hi. I am Ed Stutzcage, Lead Reviewer of Chapter 12.

I guess we will start here with Section 12.1. 12.1, more or less, just provides the general high-level design information on the ALARA design of the plant, information like the discussion of clean systems being separated from contaminated systems, that type of thing, and COL items for the COL applicant to describe their ALARA program. We reviewed that, and we have no open items in 12.1.

Let's, then, go on to 12.2. Okay. 12.2 describes the radiation sources like the RCS tanks, filters, demineralizers, reverse osmosis package, spent fuel pool, refueling pool, irradiated components, piping systems, and such.

The source terms for the shielding and radiation zoning are based on the .25-percent failed fuel percentage. And just to add on to a little bit of the discussion we had early on, yes, the .25-

percent failed fuel percentage is meant to be a conservative design basis source term. I would point out that the tech specs, if you looked at the dose-equivalent iodine, the .25 percent is actually equivalent to or lower in most PWRs; in this case, it is a little bit lower than what you would correlate to this .25-percent failed fuel. So, they can actually exceed that with their tech specs, although we would never expect them to. Normally, they're not. There have been instances of some failed fuel in plants and fairly significant, but not to the .25-percent level, at least in our knowledge, in recent years.

MEMBER MARCH-LEUBA: While you are talking about this subject, when you say a quarter of percent failed fuel, do you mean all of the fission products have left the cladding or what do you assume?

MR. STUTZCAGE: It is essentially assuming that .25 percent of the core inventory is in the RCS.

MEMBER MARCH-LEUBA: Completely just normal gases and, then, volatile gases?

MR. STUTZCAGE: No, all of it, but it does get removed by the CVCS system as it circulates through and stuff.

MEMBER MARCH-LEUBA: But, basically, assume the pellets just fell down from the fuel?

MR. STUTZCAGE: Essentially, yes.

MEMBER MARCH-LEUBA: Okay. Thanks. That's very conservative.

MR. STUTZCAGE: Yes, yes, yes.

Okay. So, that's that slide. Next slide, please.

Okay. (Making a lot of noise.) Sorry, my bad.

MEMBER SKILLMAN: I hate to use his name, but Charles will come and get you if you ever do that again.

(Laughter.)

MR. STUTZCAGE: Yes, yes, I'm sorry about that.

Okay. This is facts about airborne activity. We evaluate there is airborne activity in the containment building, the reactor building, the auxiliary building, the compound building. Again, they are based on the .25-percent failed fuel percentage. And the airborne activity is based on projected leak rates from pipes and valves and flanges, and stuff.

Go on to the next slide.

Then, we also review the accident sources, which accident sources include the accident sources for the plant operators, the main control room, filtered dose in the main control room, and the vital area access we review. Those sources are the shutdown cooling system, safety injection system, containment spray system, and, also, the main control room emergency filter.

Next slide, please.

MEMBER SKILLMAN: Edward, I'll ask it now. I don't do this stuff. So, I am honestly just trying to get educated.

When you say "accident," and, in particular, when you say "accident" in NUREG-0737, is that a design basis accident with some assumed failed fuel prior to core damage or is that post-core-damage?

MR. STUTZCAGE: For the design basis accident, it is post-fuel-damage. The guidance of 1.183 is followed, and that gives the release fractions and stuff.

MEMBER SKILLMAN: So, it is, to be very clear, that is core melt, core on the floor? Okay. Thank you. Thank you.

MR. STUTZCAGE: Next slide. Yes, slide 8.

So, in 12.2, we reviewed the radiation sources provided by the Applicant and the methodology. And with the exception of the open items discussed in the next few slides, we found the source terms to be acceptable.

Next slide, please.

Okay. On this slide, the first bullet was associated with Question 12.22. In this RAI we requested the Applicant provide missing source terms and source term details which were initially not included in the DCD. The Applicant did provide the information, provided additional information describing how they developed their sources.

In correcting some of the source terms, the Applicant didn't properly account for barium-127m. Initially, it was the only of the daughter radionuclides in the components downstream of the RCS that was considered. When revising the source terms, they initially forgot to include it. So, that was an open item in the phase 2 SER. They have come in since then and included that information, but it is still under review, that issue. We haven't got a chance to look through --

MEMBER MARCH-LEUBA: Well, you bring this item. Sorry to interrupt. I'm bringing something completely different.

The Applicant is only required to use approved COLs and methods when it affects a setpoint or something isn't in the tech specs, basically. So, they are not required to use, I assume they are not required to use approved COLs for this calculation?

MR. STUTZCAGE: No, they aren't.

MEMBER MARCH-LEUBA: Okay, but did you follow some process to verify that the COLs were acceptable? Obviously, their COL for volume was not doing quite well.

MR. STUTZCAGE: Right. These sources, the sources for the individual components were based off of -- it all starts with the RCS source term which was developed using their DAMSAM code. And then, the remaining sources, everything downstream of that, essentially, except for the radwaste systems, was done with the SHIELD-APR code, which did not account for the daughter products. But, in order to account for barium-137m, they just made it the same as cesium-137, which in reality it should be about 95 percent of the decay of cesium-137 goes to barium-137m. So, for this one, for barium-137m, it is just kind of artificially made conservative.

MEMBER MARCH-LEUBA: So, it was an interface between two COLs that was an issue or was an input --

MR. STUTZCAGE: It's the code. The code itself is limited in that it doesn't account for that. And as discussed in the following open items, we are evaluating their explanation for why it is acceptable that it doesn't account for the daughters, because of other conservatism in the code such as the five cycles of operation assumed in developing the RCS source term.

MEMBER MARCH-LEUBA: Okay. Thank you.

MR. STUTZCAGE: Yes.

MEMBER SKILLMAN: Ed, let me follow up. The gentleman behind me was explaining that, at least in part, you take ORIGEN and, as a result of ORIGEN, you I guess marry up and you end up with this Westinghouse code, and you use that code to take a look at the code that the Koreans used.

MR. STUTZCAGE: Yes.

MEMBER SKILLMAN: In at least my experience with ORIGEN, it is very thorough and it identifies all of these nuclides. So, I'm curious why there is this open item.

MR. STUTZCAGE: Well, because the APR1400, I mean they didn't use ORIGEN in their source terms for Chapter 12. So, they used DAMSAM and, then, that is where some of the changes --

MEMBER SKILLMAN: So, it's really in DAMSAM where this wrinkle shows up?

MR. STUTZCAGE: It's really in SHIELD-APR code. But, again, their approach is different than, for example, what other designs have done.

MR. BURKHART: This is Larry Burkhart. I'm the Branch Chief for Radiation Protection and Accident Consequence.

Their codes of record for the design certification will be DAMSAM and APR-SHIELD. And a lot of what we discussed was were the combination of those codes conservative. We asked them to explain why it is conservative, and they decided to do a benchmarking using ORIGEN and some of the other Westinghouse codes to show that the DAMSAM and APR-SHIELD code together were conservative. But the DAMSAM and APR-SHIELD code will be the codes of reference for the design certification.

MEMBER SKILLMAN: And are you convinced

they are conservative?

MR. BURKHART: We are still reviewing that. I could just say, from the meetings I've been in, there has been a lot of work on this benchmarking that, in my opinion, does show conservatisms, but we are still evaluating that.

MEMBER SKILLMAN: Okay. Thank you, Larry. Thank you.

MEMBER REMPE: So, while we've got you sidetracked, earlier when I was discussing this issue, I mentioned this NUREG-0409 that was used for assumptions related to the form of iodine in the RCS. And that's like a seventies vintage NUREG. I'm just wondering, and the Applicant justifiably said, "Well, that's what the NRC said to use and we were just going with their assumptions."

What gives you confidence after all these years that NUREG has given you a good number?

MR. STUTZCAGE: I think we're talking about the partition factors to the airborne activity? Is that what we're --

MEMBER REMPE: Well, it would be, according to what you have in your SER, eventually, you are using it to airborne activity, but, basically, the actual NUREG is talking about what is actually in the hot liquid that would become airborne. And it was based on this post- -- I have forgotten the other author of this NUREG --

MR. STUTZCAGE: Right.

MEMBER REMPE: -- but it is a pretty old one. There has been other work since then learned about iodine, and I'm just surprised that that would be the reference that the staff would accept. And I was curious about that.

MR. STUTZCAGE: Right. Well, we have to review what the Applicant provides. For what is in the RCS and in the fluids, that is all based on the DAMSAM and the SHIELD-APR code. But, for what gets in the airborne activity, again, they use this NUREG.

What we did is we looked at that. We looked at a couple of documents, an EPRI document. And I can't give you an extremely detailed, thorough answer off the top of my head, but, you know, the iodine is based on the iodine concentration, the temperature, the pH of the water, how much was airborne. It is based on all of that.

MEMBER REMPE: Well, this is basically what the form of iodine is. If you don't mind, I sure would like -- it's my own education again --

but what data, if you have looked at other documents --

MR. STUTZCAGE: Yes.

MEMBER REMPE: -- that gave you confidence that it is acceptable to go with 1978 assumptions.

MR. STUTZCAGE: Right. I can get back to you and give you more specifics on how we came to that conclusion. But we did conclude that, based on the data we looked at, for the temperatures that the water is, for the partition factors, that it was acceptable.

MEMBER REMPE: Okay.

MR. STUTZCAGE: But I can get you --

MEMBER REMPE: I would like to have --

MR. STUTZCAGE: No problem.

MEMBER REMPE: -- some research done on that.

CHAIR BALLINGER: That EPRI document is EPRI 3002005404, which we tried to get --

MR. STUTZCAGE: Is it?

CHAIR BALLINGER: -- which we couldn't get. I think that is the one. It is called "Advanced Nuclear Technology: Reactor Coolant Radiological Source Terms for Normal Operation -

Updated and Revised Methodology". Did you have that document? Because we tried to get it and we couldn't get it.

MR. STUTZCAGE: That is the one that you're asking about? I do not have that one.

CHAIR BALLINGER: Okay.

MR. STUTZCAGE: And I have not reviewed that document. But I will get back to you with better, more detailed explanation.

MEMBER REMPE: Thank you.

MR. STUTZCAGE: Yes.

MR. BURKHART: This is Larry Burkhart again, if you don't mind.

We do have some instances in many areas where we have Regulatory Guides or NUREGs that are old. We will get back to you on that information. We do look at, as what Ed Said, we look at what the Applicant provides us. And really, our concern is, are their assumptions conservative? So, that is what we will get to back to you about.

MEMBER REMPE: That is what I would like to know, is why you --

MR. BURKHART: Why is it conservative? There are many instances where there is new information and, for whatever reason, an applicant may not choose to pursue using newer information, or sometimes we will put a different NUREG out. But we will get back to you on that.

MEMBER REMPE: Thanks.

MEMBER MARCH-LEUBA: Anyway, so I keep getting in trouble all the time by not following procedure. You don't sent it to us; you send it to Derek or Chris.

MR. STUTZCAGE: Okay. Great. Thank you.

MEMBER MARCH-LEUBA: Don't get us in trouble.

(Laughter.)

MR. STUTZCAGE: I'll send it to Getachew, and we will make sure we get it to the right place.

MEMBER MARCH-LEUBA: It is okay to cc to us, but --

MR. STUTZCAGE: Right.

MEMBER MARCH-LEUBA: -- it goes to him.

MR. STUTZCAGE: Okay. Okay. Going on to the second bullet on this slide, that was associated with Question 12.2.16, where we requested the Applicant provide post-accident source term information in order to demonstrate compliance with the TMI requirements for accessing vital areas in GDC 19.

The Applicant provided the information. However, in Rev. 1 of the response, the accident main control room filter source term was updated, but the revised source term had an error. And supplicant to the phase 2 SER, the Applicant provided Revision 2 of the response which corrected this discrepancy and made changes to the post-accident mission doses. However, during a SHIELD outage, it was determined that the Applicant uses an erroneous post-accident recirculating fluid source term that contained an approximately 2-percent error, a small error, which we are waiting on them to, hopefully, update the response to correct and do those things. This RAI remains open and awaiting the Applicant's revision.

MEMBER KIRCHNER: So, may I ask -- these are difficult calculations to do with a high degree of fidelity unless you invest an enormous amount of effort and use a code like MCNP, and et cetera, et cetera. What uncertainty bands do you use? Since they are skirting the 5 rem --

MR. STUTZCAGE: Right.

MEMBER KIRCHNER: -- how comfortable are

you with that? And what is, in the rad-protection business in the area of shielding of personnel and dose and exposure, what is a typical band that you would put on these kinds of calculations and estimates?

MR. STUTZCAGE: Right. I mean, I don't think we have a hard-and-fast number band. But there is an open question, one of the future ones, on that, how we are so close to the 5 rem for the access to one of the areas, the vital areas, and how we can be assured that it is acceptable.

And the 2-percent error was just found as a discrepancy. It was in looking through the DCD versus their shielding, their detailed calculations, there was a discrepancy there. And we asked them about it, and they said that there is a small discrepancy. So, that is how the 2 percent came in. I guess it is the best I can answer that right now.

CHAIR BALLINGER: I mean, to expand on that, that number is given to three significant figures.

MR. STUTZCAGE: Yes, it is. Yes.

CHAIR BALLINGER: And so, the question is relevant. I mean, really?

MEMBER KIRCHNER: These kinds of

calculations, as I have said, at risk of repeating myself, are quite difficult to do with a high degree of fidelity. So, yes, that many significant digits, I would expect -- and I'm not a practitioner in this field -- that you would have some conservatisms, and I don't know -- pick a number out of the air -- because engineers count 1, 2, 5, 10, right? So, something like 5 or 10 percent or something, you know, as an uncertainty band on a calculation like this --

MR. STUTZCAGE: Right. I mean, that is partly why we have the open item. I mean, because the Applicant has provided the three significant figures in the DCD. And then, they just provided that response this week. Unfortunately, I didn't get to look at it. So, I don't have a good, a better answer than that.

MEMBER KIRCHNER: It is not so much the Applicant; it is just for you, as the reviewers, what is the expectation in terms of a confidence or uncertainty or margin? There should be some generally-applicable via the SRP acceptance criteria, right?

MR. STUTZCAGE: Well, I mean, there is no margin in the SRP or anything. And if you look at NUREG-0737, it just says it should be 5 rem.

MR. BURKHART: So, this is Larry Burkhart.

You're exactly right, we look at all the conservatisms in the calculation. We ask the Applicant to describe all the conservatisms. There is not a stated margin. I would throw out there that probably 5 to 10 percent is probably a benchmark. If we start seeing results that are 4.8 rem, 4.9 rem, we are going to ask more questions about the conservatisms, to be convinced that those conservatisms actually exist. But there is some certain amount of engineering judgment, too, that we are using to reach the reasonable assurance finding. But you're right, in the SRP there is no explicit uncertainty of 5 percent or 10 percent.

MEMBER MARCH-LEUBA: I need to talk into the microphone.

But I may be misunderstanding this. The 5-rem limit will not be exceeded in the real life because they have a dosimeter, and when you hit 5, you go home. So, the question is, these is vital operations I need to perform, like these valves? Do you have confidence that I have sufficient backup maintenance operators to send a new guy that hasn't reached the 5? Or is there something that requires that a single guy do the whole operation?

MR. STUTZCAGE: It isn't required that a single guy do the whole operation. What is required is they have to be able to show that the operation can be done with less than the 5 rem.

MEMBER MARCH-LEUBA: Okay. So, if you have backup technicians, you can --

MR. STUTZCAGE: But the operation -- excuse me; I'm sorry to interrupt -- that nobody exceeds 5 rem. So, if two people have to do it, neither of them are supposed to exceed 5 rem.

MEMBER MARCH-LEUBA: Yes, but if you send them singularly --

MR. STUTZCAGE: Right.

MEMBER MARCH-LEUBA: -- even if you will have more than 10, you will still be able to do it.

MR. STUTZCAGE: Yes.

MEMBER MARCH-LEUBA: The first one gets 5, the second one gets 5. That is the way they --

MR. STUTZCAGE: Right.

MEMBER MARCH-LEUBA: -- are doing most of the severe accident.

CHAIR BALLINGER: So, it is 5 rem per person, not 5-person-rem?

MR. STUTZCAGE: No. Yes, it is 5 rem per person.

MEMBER MARCH-LEUBA: So, that is doable, except you have to have sufficient bodies.

MR. STUTZCAGE: Right.

MR. BURKHART: And what we focus on -- this is Larry Burkhart -- what we focus on is the Applicant providing us, okay, what's the mission that this person would have to do? What's your evaluation of the rem that that person would get for the accident? But you're right, if situations -- that is why we look at the conservatisms to make sure it is somewhat reasonable. But you're right, it would be up to the eventual operator to make sure that that individual doesn't exceed 5 rem.

MR. STUTZCAGE: Yes, we look at the access path to the area, the time to perform the operation, and then, the dose on the way out as well.

MEMBER MARCH-LEUBA: Yes. So, with that in mind, I wouldn't worry too much about the third significant digit because --

MR. STUTZCAGE: Yes, I understand.

MEMBER MARCH-LEUBA: -- it has a way of self-correcting.

MR. STUTZCAGE: I understand the comment.

Okay, ready, next slide.

The first bullet is on RAI 8420, Question 12.2.22, which we asked the Applicant to provide more information on the outdoor tanks. Or, actually, this was on -- sorry -- yes, the tank levels, including the outdoor tanks.

Remaining issues include ALARA concerns associated with the outdoor tanks. Most of the questions associated with the issue have been resolved. However, the Applicant did not provide information associated with the potential dose to the public from the outdoor tanks for compliance with 40 CFR 190. In addition, the Applicant did not provide any information regarding how the tanks would be inspected. The Applicant provided a response recently that we are reviewing.

I will note that we mentioned the outdoor tanks earlier. It was discussed earlier. There is shielding information for the tanks added in a response to Question 12.2.3, I believe it is. They provided the shielding around the tanks, and they are proposing putting it in the DCD in the future revision.

MEMBER REMPE: I don't know where the place is to bring this up, but there's a lot of open items in this section and there's a lot of RAIs we were getting even this week. And maybe some of our older members can provide guidance. Is there like a limit where you say this is just not ready to bring to the ACRS? I mean, surely, I hope things won't come this way when we have to write a letter on this section, but what is the precedent? I mean, are we going to get other sections that are like this or worse? Or where is the limit in saying, oh, no, even though you've done your --

MEMBER POWERS: We've gotten a lot more than this.

MEMBER STETKAR: Oh, yes. Having been through this --

MEMBER REMPE: Okay.

MEMBER STETKAR: -- it is the whole purpose of our getting involved when we do, to see whether we, not the staff, whether we have any issues that raise a flag either for the staff or the Applicant before we get to the end of the process. So, having a large number of open items is not unusual.

MEMBER REMPE: But what if there is

content that we haven't seen yet? Are we going to write a letter on this chapter and still not have seen this?

MEMBER POWERS: We will do exactly what we have always done, Joy.

MEMBER REMPE: Which is?

MEMBER POWERS: We write letters on blocks of chapters.

MEMBER REMPE: Even if there's some major pieces that we don't --

MEMBER POWERS: And we document what we have reviewed.

MEMBER REMPE: And we ask for the additional --

MEMBER STETKAR: Yes, we flag things.

MEMBER REMPE: -- information as it is? So, there is no stopping --

MEMBER POWERS: If we want something, we can make it clear we do. I mean, you have been through this --

MEMBER REMPE: Okay.

MEMBER POWERS: -- with this application.

MEMBER REMPE: In this application, this one I think is the one that I've seen with the most so far.

MEMBER POWERS: I've seen much, much, much worse.

(Laughter.)

MEMBER REMPE: That's what I'm asking.

MEMBER STETKAR: Not on this particular one, but yesterday.

MEMBER REMPE: I just was curious. Okay.

MR. BURKHART: If I can give you some personal views -- this is Larry Burkhart -- there was a lot of discussion over the last six months about getting the information we thought we needed to write our SER with Open Items. I can say that, in my opinion, we are moving in the right direction in getting the information we need in closing out.

There were a lot of RAIs. I can just tell you that my impression is there has been a lot of work on the Applicants, then, to show us the conservatisms, and that is what we are concerned about, are there sufficient conservatisms in evaluating the radiation sources? So, I think it is moving in the right direction.

So, you shouldn't see a lot more RAIs in this area, not that you won't see any, but we are funneling-down the issues. So, I hope you won't see a lot of new content in what closes out the SER open items.

MEMBER REMPE: Yes. Thank you.

MR. STUTZCAGE: All right. The next bullet was on Question 12.2.23. That is associated with the daughter progeny issue that was discussed earlier. We are reviewing that. There is a lot of information there to review.

The third bullet was Question 12.2.25. This is kind of a similar issue with gaseous waste management system source term with the daughter progeny and, also, some other just more minor inconsistencies. That response was just recently provided, and I haven't gotten a chance to look at it yet. It was provided this week, I believe.

So, next slide.

We're going to Sections 12.3 and 12.4. So, this slide is on facility design features. Some of the design features include minimizing the cobalt content in the RCS and components in high-neutron flux areas to reduce cobalt-60.

The APR1400 has a high-efficient pre-holdup ion exchanger for cesium and rubidium, which helps reduce the dose from the outdoor tanks.

Improved reliability. A lot of the equipment is designed to last the life of the plant under reduced maintenance, such as in some areas radiation-resistant seals and gaskets and that type of thing.

By reducing leaks, for example, modulating valves or valves greater than 2 inches in diameter. Use live loading of the packing to maintain a compressive force in the packing to help reduce the leakage.

Radiation shielding is sufficient to maintain doses in walkways and frequently-accessed areas are ALARA. Most areas are less than 2.5 millirem per hour in the frequently-accessed and the walkways.

And most components with the potential for a significant dose rate are located with their own individual room which minimizes exposure during maintenance. So, you don't have a bunch of high-activity components in the same room, so you are getting dose from a bunch of different directions at the same time.

MEMBER MARCH-LEUBA: Sorry to interrupt before you go to the next slide. That 2.5 millirem per hour, is that normal operation?

MR. STUTZCAGE: Yes. This is all normal operation.

MEMBER MARCH-LEUBA: And it is for normally-accessed places?

MR. STUTZCAGE: That's the maximum -- yes, that is the upper bounds of the dose rate for the design basis source term.

MEMBER MARCH-LEUBA: Assuming that your .25 pellets drop down? So, the actual dose you expect is at least a factor of 10 lower, right?

MR. STUTZCAGE: It would be significantly lower, yes.

MEMBER MARCH-LEUBA: Because I have been in power plants where they tell you, when you get to this corridor, you run.

(Laughter.)

MR. STUTZCAGE: Yes, I --

MEMBER MARCH-LEUBA: Or you won't be able to make it through the morning.

(Laughter.)

And that was 20 years ago, not now. In this country, but it was 25 years ago, and those things don't happen anymore.

MR. STUTZCAGE: Right.

MEMBER MARCH-LEUBA: I've been in plants

a long time ago when the workers are sitting upstream of the monitor waiting for radon to dissipate so they can get out. And that doesn't happen anymore. So, a modern reactor should not design to those methods. It should be much lower.

MR. STUTZCAGE: Right, right. I mean, this is the design basis. This is with the shielding and zoning to the design basis is based on.

Okay. The next slide, slide 12, on shielding.

Shielding thicknesses for rooms containing significant radiation levels are provided to limit the dose to the radiation zones and ALARA, including during refueling and other anticipated operating occurrences.

Adequate shielding is provided to limit the dose to operators on the refueling machine and spent fuel pool handing machine platform, the 2.5 millirem per hour.

And adequate shielding surrounds the outdoor tanks and boric acid storage tank to limit the contact dose rate to less than .25 millirem per hour, except for possibly we are evaluating -- there are hatches on the tanks to access them. In that area, it may possibly exceed that, but we are looking at that. And the tanks are in the tank house, which further limits the dose and helps contain any leakage that occurs.

Next slide, slide 13.

Yes, this is on the are radiation monitors. As discussed by the Applicant, there are safety-related monitors in the containment upper operating area, the lower operating area. The spent fuel pool area monitors, two monitors there, are safety-related. And the area radiation monitors are in conformance with the SRP and the ANSI standard listed here.

MEMBER MARCH-LEUBA: Do they have any airborne contamination monitors or just radiation? Just the area?

MR. STUTZCAGE: I'm getting a little confused with another design in my head. No, I believe they don't have any built-in airborne monitors to monitor for the workers.

MEMBER MARCH-LEUBA: But that gets done periodically?

Go to a microphone.

MR. KANG: I'm Sangho Kang, KEPCO E&C.

I can give you these answers. We have

containment air monitor which is used to detect the leakage from the reactor coolant system inside the containment. That is the containment air monitoring we have.

MR. STUTZCAGE: Right.

MEMBER MARCH-LEUBA: Thank you.

MR. STUTZCAGE: So, it is not for the personnel dose. It is for detecting --

MEMBER MARCH-LEUBA: It is more for detecting accidents?

MR. STUTZCAGE: Right. And I think they do have a COL item that is part of their radiation protection program that, if portable airborne monitoring is needed, they can add in actual operation.

MEMBER MARCH-LEUBA: I am sure the HP personnel will do continuous swipes of every room.

MR. STUTZCAGE: Yes.

All right, next slide.

This slide is on minimizing contamination. This describes, gives the APR1400 general design principles, design objectives.

Prevent, minimize unintended contamination. Provisions of adequate and early leak detection capability. Reduction of cross-contamination. Decontamination and waste generation. Decommissioning planning. And they have COL items for the last two, operations and documentation and site radiological environmental monitoring.

So, that's about all I have to say on that, unless you want to get into specifics.

Next slide.

On the dose assessment, again, it gives the estimated dose rate, or the person-rem, person-sievert here. Most of the dose is from refueling and maintenance activities. And then, the post-accident mission dose for the continuously-occupied main control room and the non-continuously-occupied areas. Doses for the mission dose and the main control room are due to direct radiation and from the airborne activity in the areas.

Next slide.

MEMBER KIRCHNER: Before you go on --

MR. STUTZCAGE: Okay.

MEMBER KIRCHNER: -- so, we had considerable discussion about operational experience in the U.S. and driving down the exposures. And listening to my colleagues, I infer that the state-of-the-art in the U.S. might suggest a factor of 10 lower number than the number on your slide.

MEMBER STETKAR: State of practice. Yes, state of practice.

MEMBER KIRCHNER: So, philosophically, in the spirit of ALARA, is that an acceptable number? And allowing for, you know, it is a larger rated power plant, but that is only a small fraction of the answer. And they may use more people. So, they may have more exposure. But, in the spirit of ALARA, is that where you wanted to see a state-of-the-art design winding up or would you expect it to compete, so to speak, with what the current fleet is doing?

MR. STUTZCAGE: Well, I guess I can give two pieces of information. One is that during operation the requirement would be to always keep it as low as you possibly can. And the other thing is that these dose rates weren't very significantly different from what has been provided in some of the other DCD applications. I guess that is all I can really offer right now.

MR. BURKHART: This is Larry Burkhart.

I think it is a good philosophical question. When it comes to regulating, I think that is an acceptable number. We expected it would be lower, based on current practice. There is some conservatism, I'm sure, that goes into that evaluation. So, I would say, yes, that that is an acceptable answer, but we would expect on a day-to-day basis that the operations would drive that lower.

MEMBER MARCH-LEUBA: Yes, but at this point, well, No. 1, we have been told that that number is based on real operating experience. It is not a calculation. And No. 2 is, at the moment of the design, should we consider some mitigation, like make those operations easier to do, so operators don't have to spend that much time? Use remote action whenever possible, more shielding. There is a point at which maybe you can improve on that.

MR. BURKHART: This is Larry Burkhart again.

I think that is a very good question. The question is, as a regulator during the design certification process, what kind of additional requirements do we ask for? That is kind of what we struggle with with ALARA sometimes.

MEMBER POWERS: I will remind the members that the objective is to provide adequate safety, and that a continuous improvement philosophy that may pervade DOE does not pervade this agency, that we set an adequate standard and we hold the licensees to that adequate standards, which has margin build into it inherently. We are not in the business of running their plant.

MEMBER KIRCHNER: Dana, yes, I appreciate that. I just point out that ALARA in a sense is a philosophy of design and operations --

MEMBER POWERS: But if you look at in the regulations, you will see that --

MEMBER KIRCHNER: -- and mechanisms of action.

MEMBER POWERS: -- it occupies a peculiar position. If you look at how it is used in the regulations, ALARA has a peculiar position.

MEMBER KIRCHNER: Yes, agreed.

MR. STUTZCAGE: I will say that we do, and you probably see from reading the SER, we do ask quite a few questions on the ALARA design and review the DCD. So, there are a lot of design features built into the plant to help to try to reduce the dose.

MEMBER SKILLMAN: Well, I reacted the same way that you did to that number. And so, I went seeking information that would give me comfort that that number is in the ballpark for a new plant or is not in the ballpark for a new plant.

And the most recent version of NUREG-0713, in my view, provides the clarity that would suggest for a large plant that number is probably right about where it should be, given U.S. operating experience. But I would be quick to say that the modern fleet operators certainly do not see that as a goal for an annual exposure acceptable amount. A number 100th of that is what would be accepted in today's current environment. But that number is consistent with the last 25 years of integrated exposure on a 12-month basis for the Ps in this country, and the number has been dropping, by and large, year by year by year.

So, if you are interested, it is NUREG-0713, and the most recent version is the 47th Annual Report. It gives good information that would give you perhaps comfort for the 58-person-rem.

MR. STUTZCAGE: Move on? Okay. Thanks.

MEMBER SKILLMAN: Okay. Could I respond, one other comment? My colleague Jose was really digging into this idea about what the air monitors are sensing. And the real description for the radiological monitoring system is in Section 11, and it really ties down the answers to the questions that you were asking. What are the CAMs, containment air monitors? Are particular monitors looking for alpha and gamma and beta, exactly what you were asking for, but it is not described in Chapter 12. It is back in 11.

MR. STUTZCAGE: Right. Chapter 12 just discusses the area monitors.

Okay. So, the next slide. Yes.

Okay. So, for this section, we reviewed it and, with the exception of the open items, we found that they are meeting the applicable regulatory requirements, including ALARA and requirements to minimize contamination.

Next slide.

It goes through the open items. The first bullet on Question 12.3.8, this question asked the Applicant to provide missing shielding information in the DCD, which was included. In Revision 1 of the response, the Applicant proposed updating Tier 1 to include thicknesses for the volume control tank, to update the thicknesses for the volume control tank south wall which were originally less than the minimum thickness specified in Tier 2.

These changes were reviewed and found acceptable. However, as part of our shielding audit, we requested the Applicant provide additional justification for the shielding around piping areas. We also noted that there appeared to be a discrepancy in the dose conversion factors used in the piping calculations as opposed to the other sources.

They do a comparison. They basically take one pipe, do the calculation. They ratio it based on the number of pipes in the area. And in doing the different calculations, there are different dose conversion factors used. So, we are reviewing that, and the Applicant is going to provide us more information on that issue. So, that is why that RAI remains open. That is that issue.

The next one is on Question 12.3.10 which requested the Applicant to provide minimum shielding information for some of the irregularly-shaped piping areas and rooms. There's a couple of rooms that are very oblong-shaped with various different corners. They are not just square rooms or rectangular.

And from reviewing the initial DCD, you couldn't tell which walls had which thicknesses and the floor thicknesses and the ceiling thickness. So, we asked for more information on that. We got most of it, but there's still a few areas we couldn't tell. Over one of the hallways, for example, we couldn't tell what the shielding was over it.

And this is a significant piping area. It contains some of the hottest pipes in the plant going to the waste management systems and stuff. So, that is still an open item.

The next slide, please.

MEMBER KIRCHNER: Edward --

MR. STUTZCAGE: Yes?

MEMBER KIRCHNER: -- just out of curiosity, I am curious, how do you check their calculations, so to speak? You mentioned a complicated problem, many pipes in the same room. You have self-shielding and such. Do you use kind of a rule-of-thumb, tabular estimates to see if they are in the ballpark or do you actually run your own codes?

MR. STUTZCAGE: We do run MicroShield calculations if we need to. It was pretty simple to run. So, we run that a lot for the piping, and we can kind of multiply it to do simplified calculations.

MEMBER KIRCHNER: Yes.

MR. STUTZCAGE: That's usually what we do. And we find that something seems to be off or doesn't make sense, then we will ask questions.

MEMBER KIRCHNER: Okay. Thank you.

MR. STUTZCAGE: So, where are we? I think the next slide. Okay, slide 18.

The first bullet, related to Question 12.3.11, requested the Applicant to provide information on the CCW sump design and the CCW structure, which is a separate little structure on its own.

They provided information. The only thing that was missing was they described a radiation monitor that wasn't included in the DCD. And subsequent to the P2 SER, they included that monitor in Chapter 11. There is just a little discrepancy with the description of the monitor in the Chapter 11 that the Chapter 11 staff are asking them to resolve.

The next bullet is related to Question 12.3.13. Asks the Applicant to provide information on the access requirements of the gaseous waste management system charcoal delay beds, which are very high-radiation areas with the design basis source term. We re-asked if there was a need to access the room to work on any instrumentation, the humidity instrumentation or temperature instrumentation. We got that response very recently, and we didn't review it yet. So, that is that issue.

The next slide, please.

Slide 19 is on Question 12.3.26. This requested information on the in-core instrumentation and how you dispose of the in-core instrumentation and the dose in the refueling pool. The Applicant indicated that cutting of in-core instrumentation was performed underwater above a container to collect debris. And one of the initial questions was, do you have any kind of temporary filtration system or provisions to set up a temporary filtration system to clean up the refueling pool?

Subsequent to the phase 2 SER, the Applicant provided a response which informed us -- I was unaware at the time -- that there is the capability to directly connect the spent fuel pool cleaning system directly to the refueling pool, which would eliminate the need for a temporary filtration system.

We are currently evaluating that, though, because we had a couple of questions on if there were design provisions to prevent kind of overflooding the pool, since there is this interconnection capability. If you had the pools separated and accidentally suctioned from one pool and sent the water to the other pool, the balance-of-plant generator reviews that to make sure the water level doesn't go low and expose the fuel.

But we were just asking questions related to making sure that the pool won't overflow, either of the pools won't overflow. So, the Applicant is going to provide us a response on that.

The next slide.

On Question 12.3.46, the first bullet, which requested the Applicant to address discrepancies and provide information about preventing fires involving radioactive material and controlling the resulting dose to the public workers and minimize contamination, consistent with the guidance of Reg Guide 1.1.89, which not only provides information on the safety-related aspects of fire protection, but also on minimizing the release of radioactive material.

And the Applicant corrected a lot of

inaccurate and kind of inconsistent information and provided some more information on the design, like sprinkler systems over waste storage areas and such. However, we still have questions on venting, on the vent lines to the ion exchangers, the CVCS ion exchangers in particular, and where they are vented. We just want to ensure that they aren't contaminating the ventilation systems with potential liquid that could ruin them or prevent accidental spills out the ventilation pathway from overflows. So, that issue is still under review.

The next bullet, Question 12.3.49, this remains open. Because it described monitoring and access for the instrumentation calibration facility, and the initial DCD included an instrument calibration facility which contained a high enough source to be considered an irradiator under 10 CFR Part 36, we asked the Applicant how they were going to meet the Part 36 requirements. Well, we asked them how they were going to meet the Part 36 requirements. And they ended up proposing to remove the irradiator from the DCD design.

So, the irradiator is being -- the instrumentation calibration facility is being changed to a future use area, and they are adding a

COL item for the COL applicant to describe how that room will be used.

We found that acceptable because it isn't required that these high-level, high-activity calibrations be performed at the facility. However, we are waiting on the Applicant to provide a revised response that kind of changes everything.

CHAIR BALLINGER: This is, I think, if I recall again, this is a place where there was a high source that they were using, where if it was oriented in a certain way, then dose rates would be far exceeded. And they decided that, okay, we will just turn it around or do something to reorient the source. Is that what I'm --

MR. STUTZCAGE: No, I don't think it was ever oriented the wrong way. I think we may have asked a question about its orientation, and they provided the information. And per 10 CFR Part 36, if it is exceeding 500 R, I think, at a meter, then it meets the definition of an irradiator in Part 36 and they have to meet all of those requirements. And it did. And it did.

Okay. The next slide is the what? I think the last of the remaining open RAIs, related to Question 12.3.53, which is associated with the --

it asks about the 2-percent error in the post-accident recirculating fluid source term and getting that corrected, and also on the issue we kind of discussed earlier related to being so close to the 5-rem limit and how they are ensuring that their calculations are conservative enough that they wouldn't exceed 5 rem in the actual event of an accident. And that has just been provided this week as well, which I haven't begun to look at yet.

So, go on to the next slide, slide 22. If we can look at slide 22? Yes.

It is on Section 12.5, the Operational Radiation Protection Program. The Applicant indicated that the Radiation Protection and ALARA Program should conform to NEI 07-03A and NEI 07-08A, but the COL applicant is to fully describe the program, which is an acceptable approach and it is normally done. So, there were no open items in this section.

Next slide.

And this just summarizes that it is acceptable to defer the description of the programs to the COL applicant.

Next slide.

And that concludes the presentation.

Feel free to ask any questions.

CHAIR BALLINGER: Questions? Questions?

(No response.)

I think now we should try to get the --

MEMBER SKILLMAN: I do have one. There was a series of questions about the fuel handling equipment and the block to ensure that the fuel assemblies could not be raised to probably 12 or 16 feet below the work platform. And the text kind of indicated this is not a problem because we've got a mechanical device to prevent the elevator from raising the fuel assembly.

My question is, how thoroughly did you probe that? I was on an NSRB for a utility in the South where the interlocks and the bypasses were all defeated with jumper cables and there was a mechanical device that was not in place.

MR. STUTZCAGE: Okay.

MEMBER SKILLMAN: So, my question is, how thoroughly did you probe that question? That is a real problem if you're an operator and you're moving fuel.

MR. STUTZCAGE: Yes. I mean, I ensure that they describe that they have the interlock. I didn't look into that level of detail of defeating at the interlock or anything of that nature.

MEMBER SKILLMAN: Are you comfortable that the mechanical lock is welded in place or it is lock-wired and certified to be present?

MR. STUTZCAGE: I mean, I can say that they have an ITAAC that ensures that they have it.

MEMBER SKILLMAN: Okay. That's fair enough. Okay.

MR. STUTZCAGE: Yes.

MEMBER SKILLMAN: All right. Thank you. Thanks.

CHAIR BALLINGER: Others?

(No response.)

Okay. We will now go around. Are there any questions from people in the room, people who want to make a comment? Are there any people in the room who would like to make a comment?

(No response.)

Hearing none, shall we get the phone line open?

MR. T. BROWN: Bridge open.

CHAIR BALLINGER: Bridge open.

Are there any people on the bridge line that would like to make a comment?

(No response.)

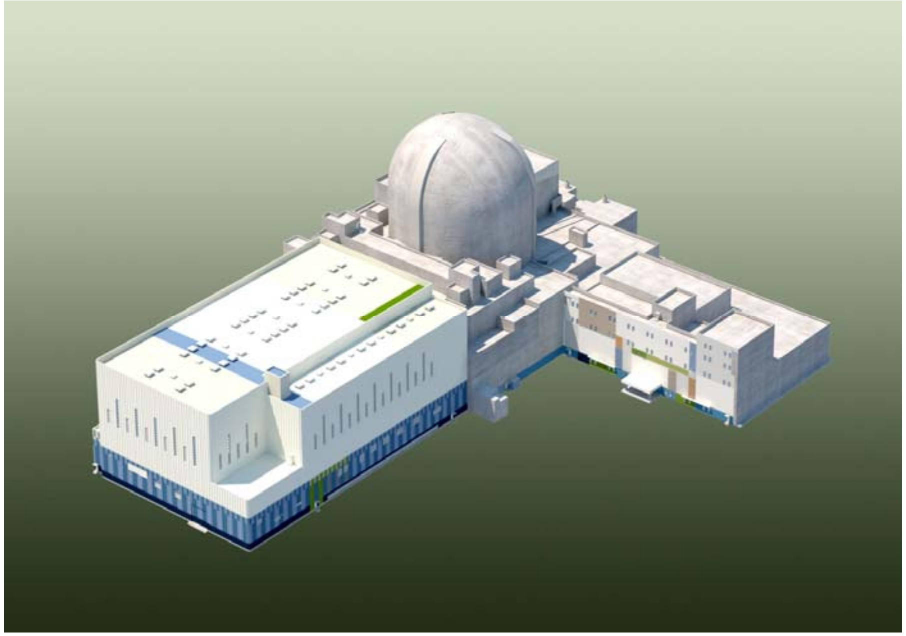
Okay. Thank you.

I appreciate, I'm sure everybody on the Committee much appreciates the presentation as well as from the Korean side.

But, if there are no other questions, then we have finished almost more than half-an-hour early, and we are adjourned.

(Whereupon, at 11:21 a.m., the meeting was adjourned.)

APR1400 DCA
Chapter 12: Radiation Protection



KEPCO/KHNP
February 24, 2017

NON-PROPRIETARY

Contents

NON-PROPRIETARY

Overview of Chapter 12 (1/2)

❑ List of Submitted Documents

Document No.	Title	Revision	Type	ADAMS Accession No.
APR1400-K-X-FS-14002-P/NP	APR1400 Design Control Document Tier 2: Chapter 12 Radioactive Waste Management	0	DCD	ML15006A039
APR1400-K-X-FS-14001-P/NP	APR1400 Design Control Document Tier 1	0	DCD	ML15006A051

❑ RAI Summary

(As of 2/23/2017)

No. of Questions	No. of Responses	Not Responded	No. of OI
83	83	0	14

Overview of Chapter 12 (2/2)

❑ Section Overview

Section	Title	Presenter
12.1	**Ensuring that ORE are ALARA**	Sang-Ho Kang
12.2	**Radiation Sources**	Sang-Ho Kang
12.2.1	NSSS Source Term (Reactor core, RCS, Spent Fuel, CVCS, SCS)	
	BOP Source Term (SGBDS, CPS, CCWs, SFPCCS, LWMS, GWMS, SWMS)	
12.2.2	Airborne Source	
12.2.3	Accident Sources	
12.3	**Radiation Protection Design Features**	Sang-Ho Kang
12.3.1	Facility Design Features	
12.3.2	Shielding	
12.3.3	Ventilation	
12.3.4	Area Radiation and Airborne Radioactivity Monitoring Instrumentation	Joon-Kon Kim
12.3.5	Dose Assessment	Sang-Ho Kang
12.4	**Dose Assessment & Minimization of Contamination**	Sang-Ho Kang
12.4.1	Dose Assessment	
12.4.2	Minimization of Contamination and Radioactive Waste Generation	
12.5	**Operational Radiation Protection Program**	Sang-Ho Kang

1. Ensuring that ORE are ALARA (12.1)

1.1 Policy, Design and Operational Considerations

NON-PROPRIETARY

1.1 Policy, Design, Operational Considerations

❑ Policy Considerations

- APR1400 provides organizational structure to implement the radiation protection policy, training, and reviews
- Applicable guidance : RG 1.8, 1.33, 8.8, and 8.10

❑ Design Considerations

- Design procedures
  - ALARA design guide provides design guidance and implementation methods
  - Incorporates lessons learned from earlier nuclear plants
  - Training programs
- Equipment design
  - Removal of contamination
  - Reduction of maintenance
  - Minimization of corrosion
- Facility layout design
  - Separation
  - Space for maintenance

❑ Operational Considerations

- COL items

2. Radiation Sources (12.2)

2.1 NSSS Source Term

2.2 Auxiliary System Source Term

2.3 Airborne Sources

2.4 Accident Sources

2.5 Key Review Items

2.1 NSSS Source Term

❑ Reactor Core

- Primary radiation : Neutrons and Gamma
- Core fission products : Calculated based on 102% thermal power

❑ Reactor Coolant System (RCS)

- Sources: Fission products, Activation and Corrosion products
- Calculated based on 0.25% fuel defect using DAMSAM code
- N-16 is the predominant activity in primary coolant inside containment

❑ Spent Fuel

- Predominant source in containment during refueling and in spent fuel pool
- 100-hour decay is ensured before movement from the core by Tech. Spec. LCO

❑ Chemical and Volume Control System (CVCS)

- Most significant sources in Aux. Bldg.
- Calculated using SHIELD-APR code

❑ Shutdown Cooling System (SCS)

- 4-hour decay after reactor shutdown is assumed

2.2 Auxiliary System Source Term

- **Secondary Systems (MSS, SGBDS, CPS)**
  - Calculated assuming SG tube leak rate of 3,270 L/day (0.6 gal/min)
- **Component Cooling Water System (CCWS)**
  - Assumed all RCS unidentified leakage (0.5 gpm for 1 hour) is transferred to CCWS
- **Spent Fuel Pool Cooling & Cleanup System (SFPCCS)**
  - Initial SFP water source is determined based on 48-hour SCS operation
  - Component sources are determined at maximum time during SFPCCS operation
- **Liquid Waste Management System (LWMS)**
  - Calculated based on expected flow rates and concentration of in-flows to LWMS
  - Use DIJESTER code
- **Gaseous Waste Management System (GWMS)**
  - Gaseous in-flows from CVCS are assumed to build up for maximum time of delay beds
- **Solid Waste Management System (SWMS)**
  - Determined based on the radionuclide inventories of the resin and filters in the CVCS and LWMS

2.3 Airborne & Accident Sources

❑ Airborne sources

- Sources of airborne contamination
  - Leaks or vents from radioactive systems (e.g. CVCS, Radwaste, HVAC)
  - Evaporation from refueling pool & SFP
- Design calculations
  - Based on the maximum allowable leak rates of component, partitioning of nuclides and activity concentrations in the fluids
  - Determine minimum required HVAC flow rates to maintain DAC fractions ALARA in all rooms in Containment, Auxiliary, and Compound buildings

❑ Accident source terms

- Design application
  - Define adequate shielding in vital areas during post-accident
  - Define environmental conditions for equipment qualification
- Determined based on RG 1.183
- Design Targets
  - Areas requiring continuous occupancy (MCR, TSC) : < 0.15 mSv/hr averaged over 30 days
  - Areas requiring infrequent access : < 50 mSv

2.4 Key Review Items

- **RAI 8420, Question 12.02-22, 23, and 25 (Dec. 22, 2015)**
  - Description of issue
    - Provide justification for not including daughter nuclides
  - Resolutions
    - Use of DAMSAM/SHIELD-APR code system for RCS and CVCS demonstrates the conservatisms
      - Comparison with another previously NRC-approved code system that considers daughter nuclides indicates conservatism of DAMSAM/SHIELD-APR system
    - For other systems including SFPCCS, SGBDS, CPS, and GRS, the source terms considering daughter nuclides were evaluated
      - It is confirmed that the shielding design margin covers the additional contributions of the daughter nuclides

3. Radiation Protection Design Features (12.3)

3.1 Facility Design Features

3.2 Shielding & Ventilation Design

3.3 Area Radiation Monitoring System

3.4 Key Review Items

3.1 Facility Design Features

- **ALARA design principles**
  - Based on RG 8.8 and 8.10
- **Applicable design areas**
  - Plant layout
  - Equipment and system design
  - Source term control
  - Airborne contamination control
  - Radiation zoning
  - Shielding
  - Post-Accident vital area accessibility

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3.2 Shielding & Ventilation Design

❑ Shielding Design

- Design criteria : RG 8.8, 40 CFR 190, GDC 19, 10 CFR 50.34
- Shielding analysis codes
  - Primary shield : ANISN
  - Neutron/Gamma Streaming : MCNP
  - Other gamma radiation: MICROSHIELD, RUNT-G
- Outcomes
  - Radiation zone maps
  - Minimum shield thicknesses

❑ Ventilation Design

- Average airborne concentration in normally occupied areas is a small fraction of DAC in 10 CFR 20 App. B
  - Determine minimum HVAC flow rates to meet the DAC fractions using airborne source term defined in Sec. 12.2
- Maintain airflow from lower to higher contaminated areas
  - ALARA design procedures ensures the HVAC system design
- Detect the time-integrated change of the airborne radioactivity within 10 DAC-hours
  - Effluent & area radiation monitors are provided

3.3 Area Radiation Monitoring System (1/3)

- **Design Objectives and Functions**
  - Warn unusual radiological events to protect personnel from possible exposure in area
  - Monitor the post-accident radiation levels in areas
- **Design Criteria : 10 CFR 20, 10 CFR 50, 10 CFR 70, NUREG-0737, RG 1.97, and ANSI/ANS-HPSSC-6.8.1**
- **Location of radiation detectors**
  - Expected frequency of access, occupancy time, and potential radiation levels in plant work areas
  - Areas where accident access to safety-related equipment is required during post-accident conditions
  - Visible and audible alarms and readouts at MCR and local area
  - Portable radiation monitor for minimizing personnel exposure and determining the optimal route to vital areas

3.4 Area Radiation Monitoring System (2/3)

❑ System Description

- Containment upper operating area monitors
  - Detect high-range gamma radiation after design basis accident to meet the criteria of RG 1.97
- Containment lower operating area monitors
  - Monitor fuel handling accident
- Containment purge isolation (CPIAS) initiated by containment upper operating area monitors and containment lower operating area monitors to prevent the release
- Spent fuel pool area monitors
  - Initiate fuel handling area emergency ventilation (FHEVAS)
- Local alarms
- Installed at detector part (RE) and electronic part (RT) depending upon the installed location
- WARN and ALARM setpoints determined by COL applicant

NON-PROPRIETARY

3.4 Area Radiation Monitoring System (3/3)

❑ ARMS List

Description	Tag No.	Class			Range	Function
		S	SE	E	Area (mSv/hr)	
Post-accident primary sample room	RE-205	N	III	N	$10^{-3} \sim 10^{2}$	AMI
Normal primary sample room	RE-285	N	III	N	$10^{-3} \sim 10^{2}$	
Main steam and FW containment piping penetration area	RE-237 RE-238	N	II	N	$10^{0} \sim 10^{5}$	
Containment lower operating area	RE-231A RE-232B	3	I	A B	$10^{-3} \sim 10^{2}$	CPIAS, AMI
Containment upper operating area	RE-233A RE-234B	3	I	A B	$10^{1} \sim 10^{8}$	CPIAS, AMI
In-core instrument	RE-235	N	II	N	$10^{-3} \sim 10^{2}$	
Containment personnel access hatch area	RE-236	N	II	N	$10^{-3} \sim 10^{2}$	
Spent fuel pool area	RE-241A RE-242B	3	I	A B	$10^{-3} \sim 10^{2}$	FHEVAS, AMI
New fuel storage area	RE-245	N	II	N	$10^{-3} \sim 10^{2}$	
Hot machine shop	RE-293	N	III	N	$10^{-3} \sim 10^{2}$	
Radiochemistry lab	RE-257	N	III	N	$10^{-3} \sim 10^{2}$	AMI
Main control room area	RE-275	N	II	N	$10^{-3} \sim 10^{2}$	AMI
TSC area	RE-279	N	III	N	$10^{-3} \sim 10^{2}$	AMI
Truck bay area	RE-288 RE-289	N	III	N	$10^{-3} \sim 10^{2}$	
Waste drum storage area	RE-292	N	III	N	$10^{-3} \sim 10^{2}$	
Compound building dry active waste storage area	RE-284	N	III	N	$10^{-3} \sim 10^{2}$	

3.4 Key Review Items

❑ RAI 8599, Question 12.03-53 (May 25, 2016)

- Description of issue
  - Since the source term is expected to be changed due to the two issues; 1) daughter nuclide effects, 2) re-calculation of the accident source terms, the cumulative impacts on the radiation shielding, zoning, mission dose, and equipment qualification design should be provided
- Resolution
  - No impact was identified due to inclusion of daughter nuclides
  - KHNP re-performed the accident source term calculations and the vital area mission doses. The results meet the dose limit of 50 mSv

4. Dose Assessment & Minimization of Contamination (12.4)

4.1 Dose Assessment

- Occupational Radiation Exposure
- Vital Area Mission Dose

4.2 Minimization of Contamination and Radioactive Waste Generation

NON-PROPRIETARY

4.1 Occupational Radiation Exposure

- **Regulatory Guidance**
  - RG 8.19
- **ORE estimation for APR1400**
  - Basis on experience data from an operating Korean 1,000 MWe PWR
    - 10-year ORE data from 2004 to 2013 for Hanul Unit 3
  - Adjusted in proportion to thermal power
- **Estimated ORE**
  - 585 person·mSv/yr (=58.5 person·rem/yr)
    - Design enhancements not included

Category of Activity	Fraction [%]	Estimated Dose [person·mSv/yr]
Reactor operations and surveillance	6.3%	36.6
Routine maintenance	24.9%	145.6
Inservice inspection	5.9%	34.6
Special maintenance	34.0%	199.0
Waste processing	1.0%	6.0
Refueling	27.9%	163.2
Total		585.0

4.2 Vital Area Mission Doses

❑ Design requirements

- Areas requiring continuous occupancy (MCR, TSC)
  - < 50 mSv TEDE for 30 days
  - < 0.15 mSv/hr averaged over 30 days
- Areas requiring infrequent access (PASS, RCCR, etc)
  - < 50 mSv TEDE

❑ Design evaluation

- Based on RG 1.183 source term (AST)
- Dose rates, transit/stay times and shielding are taken into account

❑ Results

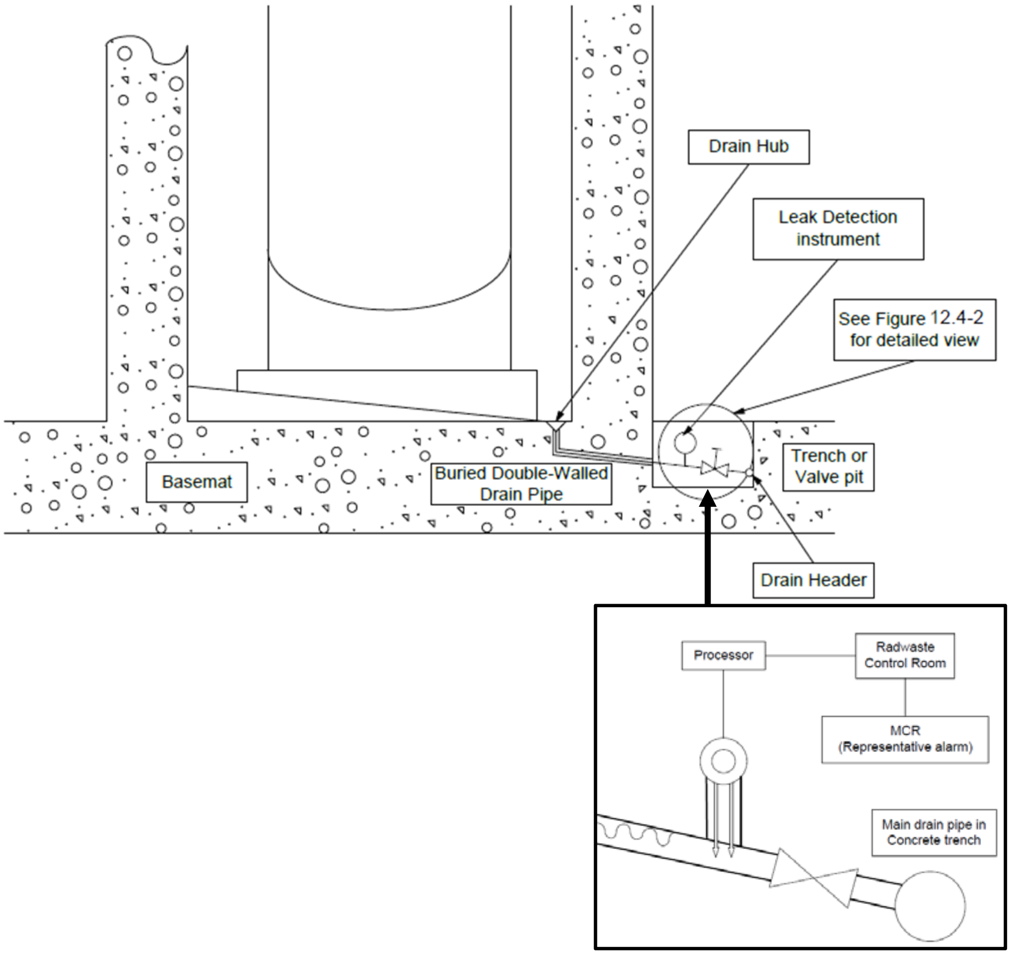
- MCR/TSC doses are within GDC 19
- Mission doses for infrequent access area meet 50 mSv

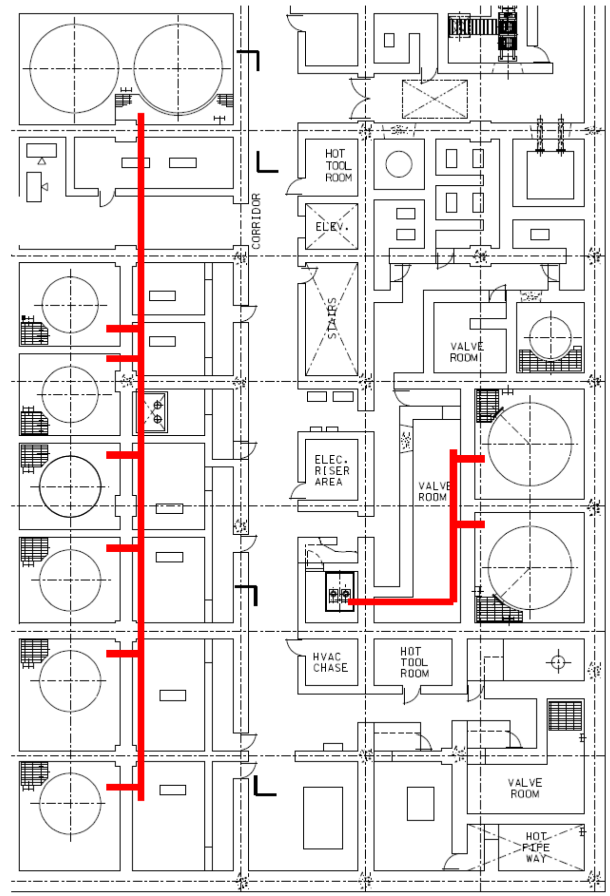
4.3 Minimization of Contamination (1/2)

- **Design requirements**
  - 10 CFR 20.1406
  - RG 4.21
- **Design/operational objectives**
  - Prevent and/or minimize contamination of the facility and environment
  - Early leak detection of an unintended release
  - Reduce cross-contamination, need for decontamination, and generation of radioactive waste
  - Decommissioning planning
  - Operations and documentation
  - Site Radiological Environmental Monitoring
- **Design evaluation & documentation**
  - Performed design review & evaluation in accordance with RG 4.21
  - Provided control measures & design features to meet RG 4.21

4.3 Minimization of Contamination (2/2)

❑ Early leak detection capability of APR1400





5. Operational Radiation Protection Program (12.5)

5. Operational Radiation Protection Program

- **Operational radiation protection program**
  - To be developed and implemented by the COL applicant to maintain occupational and public doses both below regulatory limits and ALARA
- **No open item**

NON-PROPRIETARY

6. List of COL Items for Ch. 12 (1/3)

COL No.	Description
COL 12.1(1)	The COL applicant is to provide the organizational structure to effectively implement the radiation protection policy, training, and reviews consistent with operational and maintenance requirements, while satisfying the applicable regulations and Regulatory Guides including NRC RGs 1.33, 1.8, 8.8, and 8.10.
COL 12.1(2)	The COL applicant is to describe the operational radiation protection program to provide reasonable assurance that occupational and public radiation exposures are ALARA
COL 12.1(3)	The COL applicant is to describe how the plant follows the guidance provided in NRC RGs 8.2, 8.4, 8.7, 8.9, 8.13, 8.15, 8.20, 8.25, 8.26, 8.27, 8.28, 8.29, 8.34, 8.35, 8.36, and 8.38.
COL 12.2(1)	The COL applicant is to provide any additional contained radiation sources, such as instrument calibration radiation sources, that are not identified in Subsection 12.2.1.
COL 12.3(1)	The COL applicant is to provide the material composition and shielding properties of the following doors/hatches, and these thicknesses equivalent to the minimum required concrete shield thicknesses. • Personnel Air Lock between Containment Annulus Area (100-C01) and Personnel Air Lock Entrance (100-A14A) • Personnel Air Lock between Operating Area (156-C01) and Containment Entrance Area (156-A04B) • Equipment Hatch between Operating Area (156-C01) and Equipment Hatch Access Room (156-A10A) • Door between Equipment Hatch Access Room (156-A10A) and the building exterior • Doors between Truck Bay (100-P08) and the building exterior In addition, the COL applicant is to provide the service life of these doors/hatches and perform periodic in-service inspection and maintenance for these doors/hatches to provide reasonable assurance of functionality throughout the life of the plant.

6. List of COL Items for Ch. 12 (2/3)

COL No.	Description
COL 12.3(2)	The COL applicant is to provide portable instruments and the associated training and procedures in acc ordance with 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737 as well as the g uidelines of RG 8.8.
COL 12.3(3)	The COL applicant is to establish the ARM setpoints for WARN, ALARM, and the containment purge isol ation and fuel handling area emergency ventilation actuation signals, based on the site-specific condition s and operational requirements.
COL 12.3(4)	The COL applicant is to establish how the water chemistry pH control reduces radiation fields.
COL 12.4(1)	The COL applicant is to estimate construction worker doses based on the site-specific information such as the number of operating units, distances from radiation sources, meteorological conditions, and const ruction schedule.
COL 12.4(2)	The COL applicant is to prepare a site process control program for solid waste management in accorda nce with 10 CFR61, Part 71, branch Technical Position 11-3, and other applicable regulatory requiremen ts for handling, packaging, transportation, and disposal of radioactive waste resulting from plant operatio n.
COL 12.4(3)	The COL applicant is to implement concrete tunnels for piping of the systems that may include underground piping carrying contaminated or potentially contaminated fluid to minimize buried piping. The tunnels are coated with epoxy and are equipped with sumps with liquid detection level switches. If liquid is accumulated to the detectable level, an alarm is initiated in the MCR for operator actions.
COL 12.4(4)	The COL applicant is to provide operational procedures and programs for a site radiological environmental monitoring program for the minimization of contamination control in accordance with NRC RG 4.21 and RG 4.22, as applicable, and the documentation required by 10 CFR 20.1501.
COL 12.4(5)	The COL applicant is to maintain complete documentation of system design and any site specific design modifications during the COL application, for the features for contamination control, in accordance with RG 4.21, Subsection A-3 to facilitate decommissioning

NON-PROPRIETARY

6. List of COL Items for Ch. 12 (3/3)

COL No.	Description
COL 12.4(6)	The COL applicant is to prepare a RG 4.21 Program following the guidance of NEI 08-08A. The RG 4.2 1 program shall include identification of plant-wide components, buried piping, and embedded piping, th at contain or handle radioactive materials, the built-in leak detection methods and capabilities, and the m ethods utilized for the prevention of unnecessary contamination of clean components, facility areas, and the environment.
COL 12.4(7)	The COL applicant is to prepare an offsite dose calculation manual (ODCM) in accordance with NRC R Gs 1.109, 1.111, and 1.113. The ODCM shall include a description of the methodology and parameters f or calculation of the offsite doses for the gaseous and liquid effluents. The ODCM can follow the guidanc e of NEI 07-09A for content and format.
COL 12.4(8)	The COL applicant is to prepare and implement an epoxy inspection, testing, repair, and maintenance pr ogram in accordance with RG 1.54 for Service Level I, II and III coatings. This program shall include con siderations for the design and operating objectives for implementation of NRC RG 4.21 for minimization of cross-contamination and decommissioning planning.
COL 12.4(9)	The COL applicant is to develop a leak detection program to facilitate timely identification of leaks, prom pt assessment, and appropriate responses to isolate and mitigate leakage. The leak identification progra m can be integrated into and formed part of the PCP.
COL 12.4(10)	The COL applicant is to prepare operational procedures and maintenance programs relating to the RG 4.21 features described in this system. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning.
COL 12.5(1)	The COL applicant is to provide the operational radiation protection program, including the items described in Section 12.5.

7. Open Items & Summary (1/3)

RAI No	Question No	Description	Response Submitted	Status
13-7856	12.02-02	• Not accurate Ba-137m activity in HUT, BAST, and IRWST • Not revised shielding calculation based on the revised source terms	2/22/2017	Revised response submitted
207-8247	12.02-16	• Not accurate post-accident MCR filter source term	11/01/2016	Revised response submitted (Working on additional comments)
343-8420	12.02-22	• No information on 40 CFR 190 compliance related to the outdoor tanks	01/23/2017	Revised response submitted (Working on additional comments)
343-8420	12.02-23	• Justification of the conservatism of the DAMSAM and Shield-APR code calculations	02/07/2017	Uploaded a report to ERR to justify conservatism of DAMSAM and Shield-APR codes (Under review by staff)
343-8420	12.02-25	• Significantly smaller 0.25% fuel failure waste gas dryer source than 1.0% fuel failure • Not included buildups of the daughter nuclides in the GRS source terms • Not accurate dimension of header drain tank	02/15/2017	Response submitted (Under review by staff)
141-8098	12.03-08	• Not used ICRP-51 DCF for the pipe shielding analyses • Not considered back scattering in pipe shielding analyses	01/12/2017	Revised response submitted (Working on additional comments)
141-8098	12.03-10	• Not provided many of shielding wall thicknesses	07/19/2016	Revised response submitted (Working on additional comments)

NON-PROPRIETARY

7. Open Items & Summary (2/3)

RAI No	Question No	Description	Response Submitted	Status
225-8254	12.03-11	• Provide revised response to RAI 8088 Q11.05-2 to include the CCW sump monitors and missing turbine building sump monitors	11/18/2016	Revised response of RAI 8088 Q11.05-2 Rev.2 submitted
225-8254	12.03-13	• Access control to limit radiation exposure in GRS delay bed rooms	02/21/2017	Revised response submitted (Under review by staff)
235-8275	12.03-26	• Design to use temporary filtration system during ICI cutting work	10/25/2016	Response submitted (Working on additional comments)
235-8275	12.03-43	• Inconsistencies of the information on the reactor vessel closure head vent	07/08/2016	Revised response submitted (Working on additional comments)
235-8275	12.03-46	• No description on the numerous criteria associated with fire protection of radiological material in DCD 9.5A or in the applicant's responses	12/19/2016	Revised response submitted (Working on additional comments)
376-8496	12.03-49	• Locations of alarms in truck bays and waste drum area are not consistent with ANSI/ANS-HPSSC-6.8.1.	02/15/2017	Revised response Submitted (Under review by staff)
490-8599	12.03-53	• Cumulative impacts of source term change and simplified model using MICROSHIELD code on the mission dose rate	02/21/2017	Response submitted (Under review by staff)

7. Open Items & Summary (3/3)

❑ Summary

- APR1400 radiation protection design
  - Policy and design considerations conform to associated Reg. Guides
  - Radiation sources based on 0.25% fuel defect are provided in accordance with SRP 12.2
  - Radiation protection design features are consistent with RG 8.8 to ensure ORE are ALARA
  - Estimated ORE is provided and the vital area mission doses are within the criteria in GDC 19 and NUREG-0737
  - Design to minimize contamination complies with 10 CFR 20.1406 and NRC RG 4.21
- Open items
  - 14 items are identified in staff's SER with open item
  - Most of the responses were already submitted
  - Responses for the remaining items will be provided soon

8. Acronyms

ALARA	as low as (is) reasonably achievable
AMI	Accident Monitoring instrumentation
AOO	anticipated operational occurrences
APR	advanced power reactor
BOP	balance-of-plant
CCWs	component cooling water system
CPIAS	containment purge isolation actuation signal
CPS	condensate polishing system
CPS	condensate polishing system
CVCS	chemical and volume control system
DAC	derived air concentration
FHEVAS	fuel handling area emergency ventilation actuation signal
GWMS	gaseous waste management system
HVAC	heating, ventilating, and air conditioning
LOCA	loss-of-coolant accident
LWMS	liquid waste management system

MCR	main control room
NSSS	nuclear steam supply system
OPR	optimized power reactor
ORE	occupational radiation exposure
PASS	post-accident sampling system
RCCR	remote control console room
RCS	reactor coolant system
RP	radiation protection
RPV	reactor pressure vessel
RSR	remote shutdown room
SCS	shutdown cooling system
SFP	spent fuel pool
SFPCCS	spent fuel pool cooling and cleanup system
SGBDS	steam generator blowdown system
SWMS	solid waste management system
TSC	technical support center

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Protecting People and the Environment

Presentation to the ACRS Subcommittee

Korea Electric Power Corporation

APR 1400 Design Certification Application Review

Safety Evaluation with Open Items: Chapter 12

RADIATION PROTECTION

February 24, 2017

Staff Review Team

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

- **Technical Staff**
  - Ed Stutzcage – DCD Chapter 12 Reviewer
    Radiation Protection and Accident Consequences Branch
- **Project Managers**
  - Jeff Ciocco – Lead Project Manager
  - Getachew Tesfaye – Project Manager

Technical Topics - Overview Chapter 12, Radiation Protection

- 12.1 - Ensuring that Occupational Radiation Exposures are As Low As is Reasonably Achievable (ALARA)
- 12.2 - Radiation Sources
- 12.3&4 - Radiation Protection Design Features (including Dose Assessment)
- 12.5 - Operational Radiation Protection Program

Technical Topics

Section 12.1 - Ensuring that Occupational Exposures are ALARA

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Technical Topics Reviewed:

- ALARA considerations applied during initial design
- Equipment design considerations for ALARA
- Facility layout considerations to maintain exposures ALARA

Findings:

- Based on the information supplied by the applicant, the staff determined that the general APR1400 design features and commitments are acceptable. The COL applicant will address the policy and operational considerations.

Technical Topics
Section 12.2 – Radiation Sources

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Technical Topics – Contained Sources:

- Types of contained sources
  - Reactor and Reactor Coolant System
  - Tanks and pools
  - Equipment concentrating activity
    - Filters and resin demineralizers
    - Boric Acid Concentrator
    - Reverse Osmosis Package in Liquid Waste Management System
  - Irradiated components
- Basis for stated content
  - Source terms are based on an assumed 0.25% failed fuel fraction and are used as the basis for plant radiation shielding and zoning (in Section 12.3-12.4).

Technical Topics
Section 12.2 – Radiation Sources

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Technical Topics – Airborne Activity:

- Areas potentially containing airborne activity
  - Containment Building
  - Radiological portions of:
    - Reactor Building
    - Auxiliary Building
    - Compound Building
- Basis for stated content
  - Source terms are based on an assumed 0.25% failed fuel fraction and are used as the basis for the ventilation system design.

Technical Topics

Section 12.2 – Radiation Sources

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<u>Technical Topics – Accident Sources:</u>

- Systems that recirculate fluid to cool the core during an accident
  - Shutdown Cooling System
  - Safety Injection System
  - Containment Spray System
- Post-Accident Airborne in Containment
- Main Control Room (MCR) Emergency Filter

Technical Topics

Section 12.2 – Radiation Sources

Findings:

- The staff reviewed the radiation sources provided by the applicant and the methodology used to develop the sources. With the exception of the remaining issues discussed on the following slides, the staff finds the list of sources and the methodology used to develop the sources to be complete and in accordance with applicable regulatory requirements (including 10 CFR 52.47(a)(5)) and SRP Section 12.2.

Technical Topics

Section 12.2 – Radiation Sources

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Remaining Issues:

- The source term information provided by the applicant did not properly account for Ba-137m activity, because the values were obviously too low as compared to Cs-137. The applicant is expected to revise the response to correct the Ba-137m activity in these source terms and to ensure that the plant shielding and zoning is based on the updated source terms.
- The applicant used an erroneous post-accident recirculating fluid source term (approximately 2% error in the source term) for determining radiation shielding and accident doses, resulting in a small non-conservative error in the calculated mission doses. Due to this issue, in combination with other uncertainties, such as uncertainty in what time after the accident the vital functions are required to be performed, it was unclear if the post-accident mission dose will remain below 5 rem, as some mission doses are near the 5 rem limit.

Technical Topics

Section 12.2 – Radiation Sources

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Remaining Issues:

- ALARA issues associated with outdoor tanks – the applicant has not provided information associated with the potential dose to the public from the outdoor tanks and how these tanks (which are surrounded by concrete) would be inspected.
- Radiation source terms downstream of the RCS do not properly consider daughter progeny (except for Ba-137m). The applicant is performing benchmark calculations using Westinghouse codes to demonstrate that the DAMSAM and Shield-APR codes, which were used in most APR 1400 source terms, include adequate conservatisms in the calculation of parent radionuclide concentrations that daughter radionuclides (other than Ba-137m) need not be considered.
- The applicant has been requested to provide information and to address apparent inconsistencies in the gaseous waste management system source term information. Including additional detail regarding the inclusion of daughter progeny.

Technical Topics – Facility Design Features:

- Source control
  - Minimizing Cobalt-60
  - High efficiency demineralizers for Cs and Rb to minimize dose from outdoor tanks
- Component specifications
  - Improving reliability
  - Reducing maintenance and leaks
- Radiation Zones
- Shielding for significant radiation sources (most significant sources are located within their own individual room).
- Doses in walkways and frequently accessed areas are ALARA. Most are below 2.5 mrem/hour.

Technical Topics

Section 12.3-12.4 – Radiation Protection Design Features (Including Dose Assessment)

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Technical Topics – Facility Design Features:

- Shielding
  - Shield thicknesses for rooms containing significant radiation levels are provided to limit the dose to the radiation zones and ALARA, including during refueling and other anticipated operating conditions.
  - Adequate shielding is provided to limit the dose to operators on refueling machine and spent fuel pool handling machine platform to 2.5 mrem/hour.
  - Adequate concrete shielding surrounds the outdoor holdup tank and boric acid storage tank to limit the contact dose rate on the side to less than 0.25 mrem/hour. In addition, the tanks are located in a tank house which further limits radiation exposure.

Technical Topics

Section 12.3-12.4 – Radiation Protection Design Features (Including Dose Assessment)

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Technical Topics – Facility Design Features:

- Area Radiation Monitors (ARMs)
  - Safety-Related ARMs – Containment upper operating area (high range), Containment operating area, and spent fuel pool area radiation monitors (2 each) are safety related. The 4 containment monitors have an emergency function of activating containment purge isolation and the 2 spent pool monitors actuate spent fuel handling area emergency ventilation. The containment upper operating area monitors are also required to monitor radiological conditions in containment during an accident.
  - ARMs are in conformance with ANSI/ANS HPSSC-6.8.1 (1981)

Technical Topics

Section 12.3-12.4 – Radiation Protection Design Features (Including Dose Assessment)

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<u>Technical Topics – Facility Design Features:</u>

- Minimization of Contamination - 10 CFR 20.1406(b)
  - Design objectives
    - Prevention/Minimization of Unintended Contamination
    - Provisions of Adequate and Early Leak Detection Capability
    - Reduction of Cross-Contamination, Decontamination, and Waste Generation
    - Decommissioning Planning
  - Programmatic Considerations
    - Operations and Documentation
    - Site Radiological Environmental Monitoring
      - COL Item 12.4(2)

Technical Topics

Section 12.3-12.4 – Radiation Protection Design Features (Including Dose Assessment)

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Technical Topics – Dose Assessment:

- APR 1400 total annual dose estimate of 0.585 person-Sievert
  - Mostly from refueling and maintenance activities.
- NUREG-0737 post accident mission doses
  - Continuously Occupied
    - MCR/TSC
  - Non-Continuously Occupied
    - Post-accident sampling system
    - Remote shutdown room and remote control console room
    - Class 1E switchgear room
    - I&C equipment room
    - Access areas outside the containment spray and shutdown cooling pump rooms

Technical Topics

Section 12.3-12.4 – Radiation Protection Design Features (Including Dose Assessment)

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Findings:

- The staff reviewed the radiation protection design features provided by the applicant. With the exception of the remaining issues discussed on the following slides, the staff finds the radiation protection design features to be in accordance with the applicable regulatory requirements, including ALARA requirements and requirements to minimize contamination.

Technical Topics

Section 12.3-12.4 – Radiation Protection Design Features (Including Dose Assessment)

Remaining Issues:

- The staff has requested the applicant to provide additional justification for the shielding provided around piping areas, including why different dose conversion factors were used for determining the dose from piping. These questions may require changes to the minimum shielding thicknesses provided. In addition, other pending RAI responses may impact the shielding provided.
- The staff has requested the applicant to provide minimum shielding information for irregularly shaped rooms with significant radiation sources. These wall thicknesses for these rooms were not clearly identified in DCD Table 12.3-4 in the initial DCD submittal or the response to an RAI. The responses and proposed DCD updates did not include the shielding thicknesses for all of the radiation shield barriers for these rooms. The applicant is expected to revise the response to provide all shield barriers for room 077-P01 in the Compound Building and rooms 068-A07A and 068-A10A in the Auxiliary Building.

Technical Topics

Section 12.3-12.4 – Radiation Protection Design Features (Including Dose Assessment)

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Remaining Issues:

- In reviewing responses related to radiation protection design features associated with the CCW structure, the staff determined that the CCW monitor description provided in the proposed DCD update does not accurately describe the monitor location. The description of the monitor location in the DCD remains an open item.
- The applicant has not provided adequate information on when access to the gaseous waste management charcoal delay beds would be required (such as if there are temperature or humidity sensors located within the room that may have to be worked on) and how the design meets the requirements of 10 CFR 20.1101(b).

Technical Topics

Section 12.3-12.4 – Radiation Protection Design Features (Including Dose Assessment)

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Remaining Issues:

- The applicant indicated that cutting of in-core instrumentation was performed under water in a container to collect debris and that there was no need for any kind of temporary filtration system to clean the refueling pool because the spent fuel pool purification system would be used to clean the refueling pool. The applicant contends that refueling pool water could be cleaned by direct connections from the refueling pool to the spent fuel pool purification system. This issue is currently under evaluation. The staff plans to discuss provisions to prevent pool draining or overflow due to this design, with the applicant.

Technical Topics

Section 12.3-12.4 – Radiation Protection Design Features (Including Dose Assessment)

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Remaining Issues:

- Regarding the Compound Building design to prevent the creation of flammable/explosive gas concentrations from ion exchange columns and resins and protection of stored waste from fires and combustibles, the applicant proposed adding information to the DCD specifying that the waste storage area will have an automatic fire detection and suppression system and is ventilated by the Compound Building ventilation system. This information is currently under evaluation.
- The access controls and design of the instrument calibration facility did not fully meet the requirements of 10 CFR Part 37. To resolve this issue, the applicant proposed removing the instrument calibration facility from the DCD and leaving the use of the room to the COL applicant. This is currently under review.

Technical Topics

Section 12.3-12.4 – Radiation Protection Design Features (Including Dose Assessment)

Remaining Issues:

- The staff requested the applicant to 1) ensure that the cumulative effects of all changes to source terms due to RAIs in Section 12.2 and the small error in the post-accident recirculating fluid source term discussed in Section 12.2 are appropriately considered in the shielding design, including the mission dose rates for access to vital areas during accidents; and 2) provide justification for why the results of the post-accident mission dose rate analysis is acceptable for the Remote Control Console Room and the Remote Shutdown Room, with the 2% error in the post-accident recirculating fluid source term, when the doses to access these areas were already near the 5 rem limit and the dose rate modeling for direct exposure was performed with the Microshield computer code, which is not as accurate as some other computer codes, such as MCNP, which is used for modeling some of the other radiation shielding in the APR1400 design. The applicant has yet to respond to this question.

Technical Topics

Section 12.5 - Operational Radiation Protection Program

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Technical Topics – Operational Radiation Protection Program:

- No Open Items
- Required to be provided by COL applicant
- Radiation Protection and ALARA Programs as described in Nuclear Energy Institute templates:
  - NEI 07-03A Generic DCD Template Guidance for Radiation Protection Program Description
  - NEI 07-08A Generic FSAR Template Guidance for Ensuring that Occupational Radiation Exposures are as Low as is Reasonably Achievable (ALARA)

Technical Topics

Section 12.5 - Operational Radiation Protection Program

<u>Findings:</u>

- The staff has determined that it is acceptable for the applicant to defer to COL applicants to provide information in this area.

Conclusion

Questions?

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10 CFR – Title 10 of the Code of Federal Regulations

ALARA – as low as is reasonably achievable
ARM – area radiation monitor
COL – combined license
DCD – Design Certification Document
FSAR – Final Safety Analysis Report
GSI – generic safety issue
MCNP – Monte Carlo N-Particle Transport Code
MCR – Main Control Room
NEI – Nuclear Engineering Institute
NUREG-0737 – "Clarification of TMI Action Plan Requirements"
RAI – request for additional information
RG – Regulatory Guide
SER – safety evaluation report
SRP – Standard Review Plan