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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)

U.S. EPR SUBCOMMITTEE

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THURSDAY

NOVEMBER 19, 2009

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ROCKVILLE, MARYLAND

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

COMMITTEE MEMBERS:

DANA A. POWERS, Chairman

SANJOY BANERJEE, Member

MICHAEL T. RYAN, Member

WILLIAM J. SHACK, Member

JOHN STETKAR, Member

ACRS STAFF PRESENT:

DEREK WIDMAYER

NEIL COLEMAN

NRC STAFF PRESENT:

CHRIS ADER, NRO/DSRA
SARA BERNAL, NRO/DUP/CHPB
THERESA CLARKE, NRO/DSRA
RICH CLEMENT, NRO/CHPB
JOSEPH COLACCINO, NRO
ROBERT DAVIS, NRO/SDE/CIB2
DON DUBE, NRO/DSRA
TIM FRYE, NRO/DCIP
ERVIN GEIGER, NRR/DSS/SSIB
PETER HEARN, NRO/DNRL/NARP
JOHN HONCHARIK, NRO/DE
FLORIAN JANSEN, NRO
JASON JENNINGS, NRO/DNRL/NARP
ROBERT KELLNER, NRO/CHPB
RON LaVERA, NRO/DSRA
SAMUEL LEE, NRO/DSRA
CHU-YU LIANG, NRO/DSRA/SRSB
GREG MAKAR, NRO/DE
MICHAEL MIERNICKI, NRO/DNRL/NARP
LENN MROWCA, NRO/DSRA/SPRA
JAY PATEL, NRO/DNRL
JEFF POEHLER, NRO/DE
DENDER REDDY, NRO/DSRA
EDWARD H. ROACH, NRO/CHPB

NRC STAFF PRESENT (Cont'd)

JOHN SEGALA, NRO

MOHAMMED SHUAIBI, NRO/DSRA

ANGELO STUBBS, NRO/DSRA

GETACHEW TESFAYE, NRO

ALSO PRESENT:

KATHLEEN BENNETT, AREVA NO

JIM CHUNG, MNES

KEVIN CONNELL, AREVA NP

ROBERT DAY, AREVA NP

EUGENE EDWARDS, AREVA NP (by teleconference)

DARRELL GARDNER, AREVA HP (by teleconference)

RICHARD GETZ, AREVA NP (by teleconference)

DAVID KOWALSKI, AREVA NP (by teleconference)

KAZ KOYAMA, AREVA NP

ED LINDSAY, AREVA NP (by teleconference)

JOE McCUMBER, AREVA NP

DAVID NOXON, AREVA NP (by teleconference)

MARTIN OWENS, AREVA NP

RONDA PEDERSON, AREVA NP

PEDRO PEREZ, AREVA NP

MARK ROYAL, AREVA NP (by teleconference)

SANDRA SLOAN, AREVA NP

VINOD SUD, AREVA NP (by teleconference)

BRIAN VANCE, AREVA NP

A G E N D A

P R O C E E D I N G S

8:32 a.m.

CHAIRMAN POWERS: We'll come to order here. This is a meeting of the Advisory Committee on Reactor Safeguards, U.S. EPR Subcommittee on Dana Powers, Chairman of the Subcommittee.

ACRS Members in attendance are Bill Shack with the CP one part, John Stetkar, I don't know if you have part of the reactors. Mike Ryan, Derek Widmayer is the -- of the ACRS staff is the Designated Federal Official with all the power in the meeting.

And we have an immigrant from the AP 1000 community joining us --

MEMBER RYAN: He's a member of this committee --

CHAIRMAN POWERS: -- Sanjoy Banerjee.

MEMBER BANERJEE: I am also a member.

CHAIRMAN POWERS: Yes, but you never show up. You're always drinking wine at some salubrious locale.

MEMBER BANERJEE: What can I say for that? You'll like to be doing that, too.

CHAIRMAN POWERS: Anyway, next week.

MEMBER BANERJEE: From?

CHAIRMAN POWERS: Yes.

The purpose of the meeting is to continue our review of the safety evaluation report with open items for the design certification document submitted by AREVA NP for the US EPR design.

So, if you are here for the AP-1000, you're certainly welcome to stay. You will be seduced in. You will never go back, but it you're intent on hearing about AP-1000, they are next door in the better conference room, the newer conference room. Not the better one, the newer one.

Today we will hear presentations and discuss Chapter 10 entitled Steam and Power Conversion System, and Chapter 12 titled Radiation Protection. I have no idea why those were two intimately related together in a single subcommittee meeting, but they are.

It just illustrates the flexibility of the subcommittee.

The subcommittee will hear presentations by and hold discussions with representatives of AREVA NP who brought us this lovely Normandy-like weather.

The NRC staff and other interested persons regarding these chapters, the subcommittee will gather relevant information today, but we will not be formulating any findings on these matters at the conclusion of today's meeting.

So, if you're looking for a final word on this, we're keeping that to ourselves.

Let's see. The rules for participation in today's meeting have been announced as part of a notice of this meeting previously published in the Federal Register. We have received no written comments or request for time to make oral statements from members of the public regarding today's meeting, but if you feel the urge to talk, all you have to do is get my attention, and we will certainly allow time for you to ask questions or discuss.

This is a subcommittee meeting, we're in the information-gathering, so I appreciate people piping in when they have something to contribute at any point in the presentation.

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 located throughout the meeting room when addressing the subcommittee.

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

I understand we also have a telephone bridge. Is that true?

I understand we have participants from AREVA NP on the line, as well as other members of the public. We request that participants on the bridge line, should they have something to contribute, please identify themselves when they speak, and otherwise, keep your telephone on mute during the meeting times when you are just listening.

We can -- do members of the subcommittee have any opening comments that they would like to make?

(No response.)

CHAIRMAN POWERS: I see no opening comments. We welcome the esteemed Professor of Energy to these, our august proceedings.

MEMBER RYAN: You'll never let me forget that.

CHAIRMAN POWERS: Until you do something else that distracts me.

I think we can now proceed with the meeting, and Mr. Tesfaye will give us some opening comments and some guidance and help us through the day.

MR. TESFAYE: Thank you, Mr. Chairman. Good morning everyone. My name is Getachew Tesfaye and I am the NRC project manager for AREVA US EPR, designs application project.

As Dr. Power suggested, today we'll continue our first presentation of the staff Safety Evaluation Report with open items. We begun our first ACRS presentation on November 3, 2009 where we presented Chapter 8, Electric Power and Chapter 2 Site Characteristics.

As we informed you on November 3rd -- on November 3, we have grouped the 19 FSAR Chapters into four groups, based on their first two review completion dates.

Today we will present Chapter 10, Steam Power and Conversion System, in the morning session, and Chapter 12, Radiation Protection in the afternoon, and that will conclude Group 1 of our first presentation and we will resume our first presentation in February 2010, with Group 2 chapters. Thank you.

CHAIRMAN POWERS: Okay. We're -- any questions for the speaker? You've already lined out my February for me.

MR. TESFAYE: Yes. And March.

CHAIRMAN POWERS: Well, March is already taken care of. Yes, you guys get no options in March.

Okay. Sandra, are you going to lead us out. Remember, the ground rules are the same. You are a new speaker. You have to give us some background.

MS. SLOAN: We will do that.

CHAIRMAN POWERS: So we know all about you. We don't need to know any more unless there's some good gossip.

MS. SLOAN: Nothing that I know of.

CHAIRMAN POWERS: We'll check with Ronda and see if -- Ronda, you let us know if there are any gossip about Sandra yet.

MS. SLOAN: Okay. So I'll get started. I'm the regulatory affairs manager for New Plants. If you don't know me, Sandra Sloan with AREVA, and what you'll see today in the presentations for both Chapters 10 and 12, in terms of format and type of content, very similar to how we presented information on November 3rd.

We'll start out talking about, at a high level in the presentations, what's the same about US EPR, compared to currently-operating PWR's in the US. We will identify those features that are unique for

US EPR, and then Kevin will go through, and we'll talk about section-by-section in the FSAR focusing on the high part -- the high points of the material in each of those sections.

And I did want to take the opportunity, since last time we spent some time particularly in Chapter 2 on the discussion of COL items. I wanted to pause for a minute and talk about that.

You may hear during the Chapter 10 or Chapter 12 presentations our speakers refer to COL items, and I wanted to make clear that those COL items, that's a terminology we use in the US EPR FSAR to identify particular pieces of work activities that are not within the scope of design certification, but have been identified as a responsibility of a COL applicant.

And typically, when we do that in the FSAR, we feel that we meet our responsibility by defining what the COL applicant needs to do, but we are not always required to talk about how the COL applicant has to satisfy that particular requirement for providing information.

I wanted to get that out on the table because, if we come to a point in the presentation where the speakers say, well, that's a COL item, I wanted you to understand that that's sort of our code for saying it's not within our scope for design certification and we may not necessarily be able to answer the questions, because that's just not what we provided in our application.

CHAIRMAN POWERS: Yes. The question is how much -- how much help are you going to give these poor guys, and we will probe that boundary as we did for the site characterization, and it will happen here, too.

And you'll just have to tell us where you drew your boundary --

MS. SLOAN: Okay.

CHAIRMAN POWERS: -- and then we'll tell you we didn't like that.

MS. SLOAN: We'll go with that.

So, with that said, I wanted to introduce Kevin Connell, who is our manager in the mechanical and plant design group for EPR who will be talking about the steam and power conversion system.

As you mentioned, there are also a couple of AREVA personnel on the phone. There are also supporting staff members here and I would just remind everybody that when you are initially called on to speak, if you would, as Dr. Powers said, just give a brief introduction of yourself and there may be times when our speakers may call on people on the phone.

So, our protocol for our speakers is they will identify the phone participant by name and ask them to introduce themselves and then respond to the question.

MR. CONNELL: Good morning. My name is Kevin Connell. As Sandra said, I'll be presenting the Chapter 10 Steam and Power Conversion Systems.

My background is, I have a Bachelor's in mechanical engineering from Purdue University. My Master's from South Carolina. I spent 27 years in the nuclear power business and the first 15 years I worked for Duke Power, and was involved with the late stages of construction and early start-up of that facility as well as a system engineer for McGuire, Catawba and Oconee Nuclear Stations.

Later, my career is involved with many of the steam generator replacement projects for starting with Duke Power and then for other utilities, Calvert Cliffs, as well as Arkansas Nuclear 1 and also Salem and Prairie Island.

CHAIRMAN POWERS: Enough with your background.

MR. CONNELL: Thank you.

And I joined the US EPR two and a half years ago in my current position as manager of mechanical plant design and plant engineering.

We also have with -- with me, two distinguished colleagues of mine, Mr. Robert Day, who is our turbine specialist, and Mr. Joe McCumber who is our emergency feedwater specialist.

And did you want to give a brief background of yours?

MR. McCUMBER: If that's okay, we can get that out of the way.

CHAIRMAN POWERS: Sure.

MR. McCUMBER: Good morning. My name is Joe McCumber. I'm the system engineer for the emergency feedwater system. I have a Bachelor of Science in nuclear engineering from Lowell Technological Institute, and I've been in the industry for about 38 years.

I started back with Stone and Webster, worked with them for about ten years, pretty much the whole time on the design, construction and start-up of North Anna. I spent two years down there during construction, during final part of the construction and also start-up.

Then I went on to Yankee Atomic. Most of my time there was working for Maine Yankee, and I was at that time a system engineer, lead system engineer, project engineer and then became manager. Several projects with them, including Appendix R.

Then I went on to Yankee Rose, you know as the engineering manager for license renewal which never came to fruition, and then turned into a project of the reactor, trying to save the reactor, and so my project was to come up with a way of sampling the reactor vessel.

And then I went on to Yankee Rose Dry Field Storage, and also decommissioning, and I decided to start all over again and I'm now working on EPR again, as a system --

CHAIRMAN POWERS: Getting in on the ground floor here.

MR. CONNELL: A circle.

CHAIRMAN POWERS: From the little, bitty reactor to the great big --

MR. DAY: My name is Bob Day. I've got a BS in nuclear engineering from Penn State, and I've spent 32 years designing and building and putting into operating power-generating plants, both nuclear and fossil, starting with Duke Energy and then into AREVA.

And I've been responsible for writing a

lot of equipment specs. I'm currently in the components group, and would be responsible for a lot of the major components.

MR. CONNELL: Okay. So, as Sandra said, we are going to start with the summary description and go through, pretty much follow the FSAR. We'll go through the turbine generator, through the main steam supply system and then we'll highlight the other features that are in 10.4, which take us through to turbine systems and then end up with the emergency feedwater system. We'll save the best for last today.

As we said, the main features of the US EPR are fundamentally the same as what we have in present operating plants. Many of those are -- I'll also highlight some of the differences, but these features are basically the same as what's in operating plants, the turbine generator, main steam, the feedwater condensate, turbine gland system, the condenser, the evacuation system associated with the condenser, as well as the circulating water and the steam generator blowdown system. The are all basically the same type of format. And we have some differences that I'll highlight throughout the presentation, starting with the single-flow high-pressure turbine and the single-flow intermediate pressure turbine. They are both in the same casing, so a little bit different than some of the operating plants today.

Stand-alone, start-up and feedwater system, we're using that instead of using our emergency feedwater system to start up.

CHAIRMAN POWERS: That's -- we are going to get into that, but that's really not a separate system. It's just a pump.

MR. CONNELL: No. That's correct. Yes.

And all our emergency feedwater pumps are motor-driven, and we'll get into some of the reliability and dependability aspects of that. And we have two redundant and diverse electrical overspeed trip systems for the turbine generator, instead of an electrical and a mechanical which normally are what's in operating systems today. But not with the new ones, of course.

Let's start with just a -- this is taken from our heat balance, but it gives you the idea of the flow just to have -- I know many of you are familiar with this, of course, but steam is generated from steam generator, enters our high-pressure turbine, exits the high-pressure turbine, goes through our moisture separator, two stages of reheat and then enters our intermediate pressure turbine.

Now, both of these are within that same casing. And it exists and feeds to three casings associated with the low-pressure turbine.

Collection in the condenser is then pumped -- condensate pumps through our low-pressure feedwater heaters into our de-aerator and from there goes to our feed system with our main feedwater pumps through the high-pressure feedwater heaters and then back to the steam generator.

This is a 3-D image of our turbine generator. Here you can see the high-pressure and intermediate pressure turbine all in the same casing and then our three low-pressure turbine casings and then our electrical generator.

This shows you a little bit better view, a cross-section of our turbine system. Steam would enter our high-pressure turbine in this location here and then exit, go through our reheat section and then enter our intermediate pressure turbine and then exit and then feed our low-pressure turbines from there.

We have a lead stop valve and control valve, of course, associated with the high-pressure and with the intermediate-pressure turbines.

For overspeed protection, as I mentioned in the beginning, we have two electrical overspeed trip systems. They are redundant and diverse. These systems are independent of the turbine governor system, and each system is designed and manufactured by a different vendor.

Each system is, in fact, installed in separate cubicles and we have separate power supplies power supplies associated with those.

MEMBER STETKAR: Kevin, I couldn't find any drawings, although somewhere they were referenced, the hydraulic trip locks and the solenoid valves. Are they identical for the two trip systems?

MR. CONNELL: Do you want to talk about that a little bit?

MR. DAY: Yes. They feed the same hydraulic trip block --

MEMBER STETKAR: Oh, this is a hydraulic trip block.

MR. DAY: Yes.

MEMBER STETKAR: So there's only one trip block with --

MR. DAY: Three solenoids.

MEMBER STETKAR: Three solenoids.

MR. DAY: It's got three different channels.

MEMBER STETKAR: It's one of these little -- little ported-flow type blocks --

MR. DAY: Yes.

MEMBER STETKAR: Oh, I didn't know --

MR. DAY: We lose power, it trips. If you lose hydraulic fluid, it trips.

MEMBER STETKAR: So each of the trip signals feeds a respective solenoid --

MR. DAY: Yes.

MEMBER STETKAR: -- channel on that trip.

MR. DAY: Yes.

MEMBER STETKAR: Oh, okay. Thank you.

MR. CONNELL: For turborotor integrity and missile -- turbine missiles, the rotor assembly is a series of welded forgings. It meets turbo missile requirements --

MEMBER SHACK: Well, I mean, really what you do is, you tell the COO that he has to show that it meets the turbine missile requirements.

MR. CONNELL: That's correct.

MEMBER SHACK: Just, you know, while you leave him do that, I mean, suppose he decided to get a true monoblock rotor instead of a series of welded forgings.

MR. CONNELL: Then he would have to file a departure at that -- and justify it and then come up with a turbine missile analysis that would be associated with that.

MS. SLOAN: And I think that's trying to -- a reflection of the fact that some of this information is available when you actually choose a vendor and you work that out through the procurement process.

MEMBER SHACK: I am just trying to figure out, you know, as Dana said, you know, where you stop, you know, you tell him to get welded forgings rather than a true monoblock, and then you let him do the rest.

Why not just tell him he gets a turbine that meets this?

MS. SLOAN: I think it's part of the classical -- part of Part 52. I mean, you want to specify enough to make a safety determination so the staff can make that safety determination, but yet leave the COL applicant the flexibility they need when they procure the components with the different vendors.

MR. CONNELL: The materials that we specified would be vacuum melded degassed nickel chrome moly alloys, and again the COL applicant will procure a turbine that meets or exceeds the FSAR bounding specs or provide suitable justification for the departure.

Now we move on to the main steam supply system. This is a schematic showing one of four divisions of the main steam supply. Here we have steam exiting the steam generator. We have our main steam relief tree here. I'll go into a little bit more detail on this.

We have two main steam safety valves. We have main steam isolation valve as well as the warming line, bypass line, and then it exits the safeguards building and it goes across the pipe bridge to the turbine.

MEMBER SHACK: On this one, too, I mean you specify A106 as the materials for the main steam and the feedwater lines. You do give a .1 chromium --

MR. CONNELL: Correct.

MEMBER SHACK: -- minimum. For a 60-year design life, why that rather than a higher allow, you know, P11?

MR. CONNELL: Well, we feel that's sufficient, along with their inspection program to determine whether or not --

MEMBER SHACK: Yes, but he's got to

inspect for 60 years whereas you only have to weld once. I mean, it makes your life more difficult to do the low-allow welding, but it's all done.

MR. CONNELL: Well, and we could always go even more than that, but we wanted to specify something for our customer, and if the customer chooses to have that and then have the inspection program that goes with it or go with the higher alloy, and then maybe have a little bit less of a risk of future -- for future replacement, then that could be it.

But we wanted to, again, specify what we think were minimum requirements.

MEMBER SHACK: They certainly seem like minimum requirements, yes.

CHAIRMAN POWERS: I mean, what -- you are building a couple of these plants. What are they using there? Are they meeting these minimum requirements or --

MR. CONNELL: Yes. Yes.

MEMBER SHACK: And the customer hasn't asked for a little more margin?

MR. CONNELL: Not generally. Bob, do you know if any of --

MR. DAY: No.

MR. CONNELL: -- the specifications that are more stringent than this?

MR. DAY: They have not specified that they would like to use the one and a quarter.

MR. CONNELL: How about the UP design also?

MR. DAY: No. They are using a -- well, they are using a French material that --

CHAIRMAN POWERS: Well, okay.

MEMBER SHACK: How much chrome does it have?

MR. DAY: It doesn't have the tenth of a percent chrome. It's their version of carbon steel. It's their --

MEMBER SHACK: It doesn't really even have a minimum on the chrome, then?

MR. DAY: No.

MR. CONNELL: Let me first highlight the -- what we call the main steam relief train. This consists of a main steam relief isolation valve, and then a control valve and then the silencer associated with that.

The capacity of this relief train is 50 percent of full power of each of these main steam lines, and the set pressure is 1370 psig. Now, design pressure for this is 1435, so we set the pressure down so that we wouldn't challenge the safety's on this. This -- yes, go ahead.

MEMBER STETKAR: Yes. Go back there. Is it fair game to ask you in this subcommittee meeting about the operation of the main steam relief control valves, the cool-down functions and things like that?

I noticed you're going to skip over this stuff really quickly, so this is the only opportunity to ask you about that.

MR. CONNELL: No this -- Bob, did you want to go into any detail of that?

MR. DAY: On the control valve it -- when you first --

MEMBER STETKAR: Before I ask you about the operation, I want to make sure I understand the capacity correctly because there seemed to be a bit of a difference in a couple of numbers that I read.

Each of the relief valves can relieve, as you say on the slide here, 50 percent of full flow, full rated flow from the respective steam generator, right?

So, a single relief valve, if it's fully open can relieve about 13.75 percent of full-rated core power, is that correct?

MR. CONNELL: If you're -- yes.

MEMBER STETKAR: If I take 50 percent of 25 percent --

MR. CONNELL: I just haven't done the math yet, but it sounds --

MEMBER STETKAR: It's 12 and a half percent, or something like that. They are pretty big valves, and as I understand it there are normally -- the control valve is normally in the full open position at 100 percent power.

MR. DAY: Yes, above 50 percent. It goes to open, full open.

MEMBER STETKAR: Full open. About 50 percent plant power?

MR. DAY: Yes.

MEMBER STETKAR: Okay. So, it's sitting there full open, and when the -- when the isolation valve opens, the control valve then must throttle closed to maintain pressure in the steam generator at whatever the set point is.

That's a little bit backwards from most systems that I'm used to where the isolation valve is normally open and on increasing pressure the control valve comes open.

What's been your -- do you have this

system installed in any operating plants configured this way?

MEMBER SHACK: And what's been your --

MEMBER STETKAR: Overcooling transients. It seems that this design might be susceptible to excess steam flow.

MR. CONNELL: Well, it's been modeled and analyzed in Chapter 15, of course, but OL3, do you want to speak a little bit to that, Bob?

MR. DAY: Well, we haven't even started up yet, and this may have come out of the M4 designs in France.

MEMBER STETKAR: I mean, do you know of any French plants -- I'm not familiar with the French plants. I'm familiar with the German plants and the German plants where they have the -- whatever you want to call it, the program cool-down mode, keep the control valve closed and the isolation valve open and throttle control valve open.

MR. DAY: I believe the reason that this way is, the isolation valve will seal a lot better if it's closed. Control valves aren't made to isolate at normal. So, you're going to have potentially more leakage if you try to use the control valve.

MEMBER STETKAR: Yes, although --

MR. DAY: It's the backup --

MEMBER STETKAR: -- the vast majority of the plants in the world operate with an open isolation valve and accept the minor leakage if any from coolant.

I was just curious whether there was any operating experience with this type of configuration to indicate whether there is a propensity for overcooling of transients.

In other words, you know, overcooling and then subsequent isolation of that division or all four divisions with a challenge to the safety valves.

MR. CONNELL: Well, we can take that as a question.

MR. DAY: The control valve is going to control based on the main steam air pressure.

MEMBER STETKAR: It's going to control as long as it doesn't stick.

MS. SLOAN: I think what you're hearing to say we don't exactly know the answer for the European plants.

MEMBER STETKAR: I would be curious. If you could get it -- I mean, it's not necessary a safety issue except for challenges to overcooling transients which have different implications on subsequent plant response.

MR. TESFAYE: Tim Stack.

MS. SLOAN: Tim, do you want to get a microphone -- is there a microphone there for you, Tim? And introduce yourself.

MR. TESFAYE: Okay. My name is Tim Stack from AREVA. I'm the manager of technical integration with AREVA in our products and technology group.

My background, I have a Bachelor's and a Master's degree from Penn State in Mechanical Engineering. I've worked for AREVA and previously Frametone and B&W since 1982. My background is basically in NSSS system design as well as turbine site design and BOD design, so I've covered pretty much the whole plant.

Recently, before the EPR, I've worked on power operates and steam generator replacements predominantly, but I've also worked in many of the plant programs in the past, fire protection, EQ. And that's basically what my background is.

Relative to this question, as you indicated, basically it's kind of the opposite way that you would normally see an atmospheric dump valve, if you wish, in an operating plant.

And realistically, the reason behind it

was to get the -- really the desired plant response that you want. If you did it the normal way you're basically going to have a slow response in the control valve because it's a motor-operated valve and it would be too slow to respond.

Typically in the operating plant you'll have an AOV as the control valve and you'll have an MOV for the isolation valve, and the AOV can respond very rapidly, and that's how you -- and that's how you manage your transient.

Here, again, with the MOV as the control valve, you couldn't get the response you wanted. So, that's really the reason why basically it looks backwards.

MEMBER STETKAR: Okay. Okay. And I understand now why it's -- because it is an MOV, I understand why it's -- why it's open. I'd still like to know if any operating plants in France, for example, have this configuration and what their experience might be.

MS. SLOAN: I think we'd have to follow up on that.

MEMBER STETKAR: Okay.

MEMBER BANERJEE: Can the CANDU's do something similar or control through the secondary sites?

MEMBER STETKAR: I don't know, Sanjoy. The only CANDU I'm familiar with is so old that it's irrelevant, so I don't know that can be designed.

I do know that -- I know the Konvoi and pre-Konvoi designs in Germany, and they have a faster-acting cold valve that's normally closed. They do have the rapid secondary cool-down requirements because they have the intermediate head pumps like this plant.

But, they operate with the isolation valve normally open, the control valve normally closed, and it comes open on a program pressure signal, which I'm -- you know, they are also susceptible to possible overcooling if the valve opens too far, but it's just a little different.

This one you essentially have to snatch it, if you will, in the downward direction, which the valve sticks at all, you're --

MEMBER BANERJEE: What is the rate of cool down for -- that's required of asking AREVA, of course, not. Is it within sort of the experience base that Dr. Stetkar is talking about or have we got operating experience, plants that have this type of cool down rate?

Maybe the German plants do. Is it within that rate?

MR. CONNELL: Of course, we factor that back into our Chapter 15 analysis, and it's -- and it complements our Chapter 15 analysis. How it compares to the other --

Do you have any comparison, Tim, or Bob?

MEMBER STETKAR: Well, I guess the question that Sanjoy was asking is, at least what's the design cool down rate --

MEMBER BANERJEE: Right.

MEMBER STETKAR: -- for an SI signal.

MEMBER BANERJEE: Is it within that range?

MEMBER STETKAR: In terms of degree C per hour or whatever.

MR. TESFAYE: And I'm going to going to apologize. I should double check this. I'm going from memory, but it's 180 degrees Fahrenheit per hour.

MEMBER STETKAR: Oh. Okay.

MR. TESFAYE: Fahrenheit.

MEMBER STETKAR: Fahrenheit.

MEMBER BANERJEE: So, a hundred degrees Celsius.

MR. TESFAYE: Yes. A hundred degrees Celsius.

MEMBER STETKAR: Okay. That is comparable to the other ones.

MR. CONNELL: And then, in addition to the main steam relief train, we have two main steam safety valves. Each has a 25 percent full-power main steam line flow. The first one is set at 1460 and the second one is set at 1490 psig.

Then we have the main steam isolation valve, as well as the bypass main steam warming isolation valve and steam warming control valve. The main steam isolation valve is set for a closure time of five seconds or less against a differential pressure in either direction of 1320 psid.

And again, we have the warming valves that we use during the start-up period.

CHAIRMAN POWERS: What is the leakage across that valve? What is the leakage across your main steam isolation valve?

MR. CONNELL: Leakage requirements across that -- Bob, do you know what the --

MR. DAY: It will be specified as 2 cc's per hour per inch of valve size. It's a 30-inch valve, essentially.

MR. CONNELL: Now I will go through some of the other features -- oh, before I go into that, we had talked briefly about the .1 percent chromium and I did want to mention that that's the material selection and of course the COL applicant has the ability to go more than that, and also to -- they are required to have an inspection program that addresses the wear through the life of the plant.

Main -- now I'll go through other features of the system. Most of these are plain vanilla systems associated with the turbine. Our main condenser, of course, performs no safety function. It's a multi-pressure three shell de-aerating condenser designed to HEI standards.

The COL applicant will select the materials based on the site water characteristics, mainly would be stainless steel, but could go to titanium as necessary.

Designed for turbine bypass flow of at least 50 percent of the main steam flow without tripping due to the high back pressure and the hotwells are compartmentalized to help identify any locations of the leakages.

MEMBER STETKAR: The full capacity of the turbine bypass is about -- if I can read this, 65 percent if you account for all six valves.

Does -- is the EPR designed to take a full load rejection without a reactor trip?

MR. CONNELL: Yes.

MEMBER STETKAR: It is? So you rely on a reactor run-back, whatever the EPR calls it, a rapid rod insertion to take the extra 35 percent power off the reactor?

MR. CONNELL: That's correct.

Tim, you know a little bit more about that --

MEMBER STETKAR: Initially, I mean, you know --

MR. CONNELL: Yes, initially. They talk about this in some of our previews.

MR. TESFAYE: In general what you're saying is exactly correct. If you had a load rejection from 100 percent power you would get a partial rod drop which will reduce reactor power and then you'll match it between what's within -- in the turbine as well as the turbine bypass, and if pressure continued to rise, open the MSRT's until you got to the stabilized state.

MEMBER STETKAR: Okay. I was a little -- there was a little -- there was one sentence in the FSAR that had me confused because it said subsequent to loss of external load, the turbine trips and it was followed by a reactor shutdown, so --

MR. CONNELL: It is reactor run-back, yes.

MEMBER STETKAR: It is a reactor-- but shutdown as the run-back.

MR. CONNELL: Yes.

MEMBER STETKAR: And not a trip. Okay. Thanks.

MR. CONNELL: We also have a main condenser evacuation system. It performs no safety function. It has two redundant holding vacuum pumps per condenser shell, has one main hogging pump for the whole system. We use that during start-up and exhaust from the vacuum pumps routed to the turbine building air vent system.

MEMBER STETKAR: Why did you go to -- I don't have any operating experience with mechanical vacuum pumps rather than air ejectors. Was there any -- any reason or is that just standard design in the French fleet?

MR. CONNELL: Bob, do you have any experience with that, too?

MR. DAY: Right now it's kind of standard design and --

MEMBER STETKAR: Is it really?

MR. DAY: -- in the US for current

generation of fossil plants.

MEMBER STETKAR: Okay. Oh, okay.

MR. DAY: The state of condensate vacuum pumps --

MEMBER STETKAR: Is a lot better?

MR. DAY: -- is a lot better than it used to be --

MEMBER STETKAR: Thanks.

MR. DAY: -- back when people decided a steam jet ejector was --

MEMBER STETKAR: Was the way to -- okay. Okay. Thanks. I was just curious.

MR. CONNELL: We also have the turbine gland sealing system. It performs no safety function. It collects seal leakage from the main steam stop and control valves, both the high-pressure and intermediate pressure shaft seals also.

It provides sealing steam to low-pressure turbine shaft seals. Steam and noncondensables are routed to the gland steam condenser and the air or noncondensables are exhausted to two redundant -- through two redundant exhaust fans.

Turbine bypass system. This is not a safety system, but we introduced this earlier. Sufficient capacity to provide actuation of the main steam -- to prevent actuation of the main steam relief following a turbine trip and full-load rejection.

There's six turbine --

MEMBER SHACK: I thought Tim said the main steam relief would go off. Isn't that what you just said?

MR. TESFAYE: What I said was that, in general, the way it's designed is that you would have a partial rod drop and the turbine bypass would be capable of accommodating it. If the pressure did -- if we had some other upset, the pressure did continue to rise, then the MSRT would open. So that would -- that's really a fall-back on the MSRT, but the turbine bypass is designed to accommodate that.

MR. CONNELL: And as noted earlier, we have six valves. Five of them required for the 50 percent, so we have one out for maintenance and still have the 50 percent steam flow capability.

During normal operation we start up, is bypassed to the condenser and then we use this to control our cool-down.

Circulating water system. This doesn't perform a safety function also. It supplies cooling water to the main condenser and aux cooling water systems rejects heats to the environment via the normal heat sink.

And again, we'll talk about the scope of the DC. The DC is limited to the design inside the turbine building. COL applicant will do the site-specific design, anything outside the turbine, and select the condenser materials depending on the water source.

MEMBER STETKAR: Kevin, before you go on to the condensate polishers, a couple questions on circ water. What is the auxiliary cooling water system, and where can I find a description of it in the FSAR? I did quick work searches, but I couldn't find it anywhere. So I was curious of what it is.

MR. CONNELL: Bob, do you want to --

MR. DAY: It's just the takeoff before the circ water goes into the condenser and it goes -- it provides cooling water to the turbine plant.

MEMBER STETKAR: So it's --

MR. DAY: That was the plant cooling -- closed cooling water system, heat exchangers.

MEMBER STETKAR: Okay. It's just --

MR. DAY: So it's just from -- it's actual running circ water over to cool.

MEMBER STETKAR: Okay. So it's cooling for what I would call the turbine closed cooling water system?

MR. DAY: Yes.

MEMBER STETKAR: Is it --

MR. DAY: It's not safety-related.

MEMBER STETKAR: I understand that, but there's no -- there's no actual description of the flow path or valves or anything anywhere in the FSAR.

MR. DAY: It wasn't asked about in any of the SRP's --

MS. SLOAN: Yes, I don't think it's in the SRP. It's a non-safety system.

MEMBER STETKAR: Okay.

MR. CONNELL: A non-safety system, right.

MEMBER STETKAR: Another question. I'll try to remember that. The other question, your circ water system has kind of a different flow path where you run circulating water in series through three the condenser sections which is different from most circ water systems that I've seen, which is for more parallel flow path configuration.

This means if you have a leak in any one of those condenser sections, you basically have to isolate half your main condenser, right? They are only isolation valves in the circ water side on either end.

MR. CONNELL: Yes.

MEMBER STETKAR: Which means, you know, any leak in measurable -- reasonable leak, let's say, you have to --

MR. CONNELL: That's correct.

MEMBER STETKAR: Any particular reason for that, rather than going through a parallel flow configuration?

MR. CONNELL: Well, this gives us --

MEMBER STETKAR: Is this also a standard design?

MR. CONNELL: Well, this gives us our differential pressure in our condensers. Do you want to --

MR. DAY: I guess by serious flow you're talking about into each of the shells and then out of each of each of the shells instead of --

MEMBER STETKAR: Well, the flow diagram that I saw showed -- showed the inlet -- and I might be on the wrong end, but the inlet -- the cold circulating water coming into the inlet of the low-pressure condenser, and then going -- exiting the low-pressure condenser going into the inlet of the --

MR. DAY: Yes.

MEMBER STETKAR: -- intermediate pressure condenser --

MR. DAY: Correct.

MEMBER STETKAR: -- and so forth, and finally exiting the high-pressure condenser. So, it's heated up progressively as it goes on, where most designs that I've seen have been parallel flow.

If you have six condenser sections like this, you just have six parallel flow paths.

MR. DAY: If you have a plant with direct cooling, which a lot of the existing operating plants are, then you would -- it would probably be more efficient for you to go with a single pressure condenser.

MEMBER STETKAR: Yes.

MR. DAY: Which is what that is.

MEMBER STETKAR: Yes. That's right.

MR. DAY: The reason we went with the multipressure condenser is because it will -- the heat rate works out better --

MEMBER STETKAR: Okay. That's what I was thinking about. Could you --

MR. DAY: A lot of the fossil plants built nowadays are on cooling towers, and they have --

MEMBER STETKAR: That's what I was going to ask. You probably couldn't meet EPA discharge temperatures on this plant with an open -- open cycle cooling system, could you?

MR. DAY: No.

MEMBER STETKAR: Okay. So you --

MR. DAY: You could try and get a permit, but --

MEMBER STETKAR: Okay. I understand why it's -- thanks.

MR. CONNELL: Good question. Thanks.

The condensate polishing system, this, of course, does not provide any safety function, but it removes the corrosion products impurities during start-up and keeps the water chemistry within the EPA EPRI water standards. It's design capacity is one-third full condensate flow.

That brings us to the condensate and feedwater system itself. Operationally this, of course, provides water for the steam generators at the required temperature, pressure and flow rate. It provides warm-up functions during start-up and cool-down, and that's the start-up pump that we have, the feedwater pump.

Safety functions, the condensate and feedwater system provides redundant isolation. It's mainly isolation, of course, prevents or reduces an overcooling event due to control malfunctions or line breaks.

It prevents depressurization of the steam generator by the isolation, retains any radioactivity in an affected steam generator, from a steam generator tube rupture, and then shut off of the feedwater supply to prevent containment overpressurization in case of a main steam or main feedline break.

MEMBER STETKAR: Are the -- do the full load isolation valves close automatically after every reactor trip? Again, I'm trying to figure out how the system works during transients.

MR. CONNELL: Yes. I believe it has an automatic --

MR. DAY: On the reactor trip the full load isolation would close, the low load isolation would stay open.

MEMBER STETKAR: Would stay open, yes. Okay. But the full loads, the full loads go closed. And this might be getting too far into the control systems, but how does the -- how does the feedwater control system respond to a reactor trip?

There was a brief discussion of a shut-down, that it says that there's a program shut-down where you take off one condensate pump. You take off a feedwater pump. As power goes lower, you take off the second feedwater pump and finally get down to the start-up/shut-down feedwater pump.

How does the system respond to a reactor trip? Does it just go away or does the control system limit you to only the start-up and shut-down feedwater pump?

What I'm thinking about -- I'll give you the reason for my question. I'm a PRA guy, and I think about success criteria for being able to use this equipment after a reactor trip or a turbine trip.

So, that's kind of the reasons for my answers -- or my questions, anyway.

MR. CONNELL: Yes.

MEMBER STETKAR: If you don't want to discuss, it's more appropriate to discuss it in a different subcommittee meeting -- I was just curious about how this -- this system works a little bit.

MR. CONNELL: Well, I think we can go into some of the basics of it.

MEMBER STETKAR: Not so much a program shut-down, but I'm more interested what happens if you get a reactor trip, how does it --

MR. CONNELL: You get a reactor trip, the one valve would close. You'd still have flow through the 10-inch line going to the steam generator and based on feedwater flow you would drop off at least one feed point. You'd probably get down to two.

MEMBER STETKAR: But do the feed pump -- when the feed pump trip signals come off feedwater or steam flow imbalances or steam generator level or reactor power or --

MR. CONNELL: They --

MEMBER STETKAR: I'm trying to think about what the control system is doing to me as an operator in terms of leaving me equipment available.

In other words, if the control system's automatically tripping off and locking up pumps on me, I can't use them.

MR. CONNELL: I see Tim is anxious to answer this question.

MEMBER STETKAR: Okay.

MR. TESFAYE: When you -- it's described when you look at the normal reactor trip it will isolate some of the normal feedwater flow path. That's consistent -- I'm sorry. That's consistent with the way that we've done it on some of our operating plants where we try to isolate the high-flow pathways and limit the flow to prevent, you know, an overfeed.

MEMBER STETKAR: Yes. Yes.

MR. TESFAYE: So again, that's very consistent with --

MEMBER STETKAR: Yes. No, I understand that the flow path isolation -- I'm thinking more about what it's doing to the condensate and feedwater pumps.

MR. TESFAYE: Typically when you look at even on the operating plants, as you -- you don't necessarily need to trip the plants initially. You'll just open up -- you have a high-flow recirc so that the pumps will go back on recirc as you push them back on their curve.

MEMBER STETKAR: That's on the --

MR. TESFAYE: They will stay in operation.

MEMBER STETKAR: Will they also stay in operation on the EPR?

MR. TESFAYE: Yes.

MEMBER STETKAR: That's what I'm asking. Oh, okay.

MR. TESFAYE: Okay. So what's happening on our -- for a garden variety turbine trip of normal transient our primary means of decay heat removal is main feedwater.

MEMBER STETKAR: Yes.

MR. TESFAYE: It is the preferred means.

MEMBER STETKAR: Yes. Yes.

MR. CONNELL: And we don't want to lose that.

MEMBER STETKAR: Right.

MR. CONNELL: We want to get that heat back going --

MR. TESFAYE: That is very consistent with the operating --

MEMBER STETKAR: That is, and the only reason I was asking this question was, when I read the brief two-paragraph discussion of the system operation during a plan shut down, it told me that the control system is apparently automatically tripping pumps, or at least I was led to believe that.

MR. CONNELL: We'll go back and clarify that if needed.

MEMBER STETKAR: And my question came out, well, if that's -- if there's a program logic that automatically does, indeed, shut down pumps as something is ramping down -- I don't know whether it's done on turbine power or steam flow or feed flow or whatever it is.

MR. TESFAYE: I think part of the difference is what's happening in the automation and the distributor control system relative to how does it normally start the plant up or shut it down versus --

MEMBER STETKAR: Yes.

MR. TESFAYE: -- what's happening during a transient.

MEMBER STETKAR: Yes. And what I didn't know is what happens after a transient.

MR. TESFAYE: And again, during a transient, it's very similar to the operating plants.

MEMBER STETKAR: Okay.

MR. TESFAYE: We want to maintain feed -- main feedwater availability to the extent we can.

MEMBER STETKAR: Thank you.

MR. CONNELL: All right. Next we have the steam generator blow-down system. Operationally it does what most blow-down systems do. It removes contaminants from the steam generators. Blow-down is demineralized and return to the steam and condensate systems.

It's continuously monitored for radioactivity to detect any steam generator tube leaks and it has the capability to discharge to the liquid waste system if we detect any primary to secondary leakage.

Safety functions are isolation-related

also. Isolation on emergency feedwater actuation and also on containment isolation signal it will isolate and then it isolates also the affected steam generator in case of a steam generator tube rupture.

MEMBER SHACK: What is the sensitivity of the leak detection?

MR. CONNELL: For our radiation protection monitors, I'm not sure if --

Tim, do you have any information on -- handy?

MR. TESFAYE: Handy? No, I don't. However, in this afternoon's session we are going to have -- Pedro Perez will be participating and he may be able to have that answer for you. So, we can hold that question till the afternoon?

MR. CONNELL: We can follow up this afternoon on that?

MEMBER SHACK: Yes.

MR. CONNELL: Okay. And finally, we have the emergency feedwater system. Primary function is to provide flow to the steam generators and restore and maintain the steam generator order inventory, support residual heat removal system from the RCS and cool down and depressurize the RCS to OHR entry conditions following design basis events.

This is strictly a safety system. As we said, we don't use this during start-up. We have four identical safety-related trains, and they are normally aligned to separate steam generators.

We'll go into some of the schematics here, too. I'll show some of the cross-connects also.

Major components, we have storage pool, centrifugal pump and motor and the flow control valve, level control valve, steam generator isolation valve, supply cross-connect valve and discharge cross-connect valves. And, of course, the associated instrumentation and controls.

Here is a schematic showing on train, one of four trains. Where we have the storage pool for each train feeds the emergency feedwater pump through the mini-flow check valve, through the flow control valve, level control valve, isolation valve and then to the steam generator.

And we also have two cross-connections, one on the supply side, which is normally closed, and one on the discharge side, of course, is normally closed.

MEMBER STETKAR: Kevin, before you -- you might get to it on a later slide, but I might as well ask you on the picture.

The suction side cross-connect valves are now closed?

MR. CONNELL: That's correct.

MEMBER STETKAR: Okay.

MR. CONNELL: And they are manual valves.

MEMBER STETKAR: Okay, and just take a note, because I think there's still one place that you missed in the FSAR, in Section 10.4.9.223. There's a statement that says the supply header isolation valves are maintained open.

MR. CONNELL: Okay. Well, we'll take that as --

MEMBER STETKAR: Because when I was originally reading the FSAR I stumbled across open and closed --

MR. CONNELL: We're going to change that in a later RAI response -- did we or not?

MEMBER STETKAR: Well, there are two places in the FSAR that says they're closed, and one place it says they're open, and I read through the staff's RAI's, and I understand they are normally closed now, and I just wanted to confirm that, indeed, that's the case.

MS. SLOAN: Okay. We will take that and make sure that gets fixed.

MR. CONNELL: Okay.

MEMBER STETKAR: The flow control valve is normally fully closed. Is that -- you show it closed on this drawing. Is that right?

MR. McCUMBER: The valve is normally closed, but it is not fully closed, because it actually closes on a mechanic stop.

MEMBER STETKAR: That's what -- I was a little confused because it said the mechanical stop is in there to limit maximum flow for run-out protection.

MR. CONNELL: That is the upper stop.

MEMBER STETKAR: That's the upper -- again, there is a statement in the FSAR that says the FW flow control valves are motor-operated control valves that limit EFW pump flow, ya-de, ya-de, ya-da.

The valve is positioned on its mechanical stop stand-by during normal plant operation which gave me the impression that it was at its open stop and another place it says the valves are closed.

So, this shows them closed. The valves are closed to the minimum flow --

MR. McCUMBER: It's only when a mechanical stop -- it's in a closed position stop.

MEMBER STETKAR: Okay.

MR. McCUMBER: But it leaves it open a

percentage.

MEMBER STETKAR: Okay. Got it. Thank you.

MR. CONNELL: This next slide goes through some of the special features of this. We have four separate storage pools that are provided each in a separate safeguards building.

Each of these safeguards buildings are category one tornado protected buildings. The total volume of the four pools is 411,200 gallons. Our bounding condition to meet BTP-5.4 requires 365,000 gallons. So, we more than compensate for that.

Supply and discharge cross-connect headers provide the capability for each of these EFW pumps to take suction from any of the storage pools and feed any of the steam generators, and my next slide will go through some of those cross-connect scenarios.

The EFW system does not provide the normal auxiliary feedwater functions as we noted a few times earlier because it does not provide the start-up and shut-down.

Power supply. We had to mention electrical, and Brian Gardes is here also. You might be familiar with him for two weeks ago was presenting during the Chapter 8.

Each of these emergency feedwater trains are powered from separate divisions of the emergency feedwater -- I mean, emergency power supply systems.

Each of the buildings have the 6.9 kV switch gear that powers the pump, and then the 480 volt distribution panels power the motor-operated valves.

In case of off-site power, each of the 6.9 kV buses are powered by separate diesels, emergency diesels. Essential valves, the level control valve, the discharge cross-connect and the isolation valves are also fed from emergency uninterrupted power sources, power supply.

During EDG maintenance, we have the ability to do alternate feed, as described two weeks ago in Chapter 8 where Division 1 could be supplied from two or two from one and also three to four and four to three.

In addition to the EPSS, we have two diverse SBO diesels and they can supply power to the central emergency feedwater equipment during postulated station black-out events.

Now, I'll go through a cross-feed example. I'll try to show you a little bit more about how our system operates.

Here we have all four trains shown. Here's train one, feeding steam generator one. Here's the storage pool. And in this scenario -- and then, of course, the other four trains.

In this scenario we're postulating we have maintenance on the first pump, on pump and train number one.

In train number two, we're going to postulate a single failure, loss of electrical capability, so this train is out of service.

Train three, nothing's happening to, but in train four we're postulating a main feedline break.

So, in this scenario, we would start the emergency feedwater system. We would only have trains three and four available at this point, and train three would feed the intact steam generator and train four would feed the faulted steam generator.

And in this scenario we've run through Chapter 15 analysis and we can -- this scenario is fine to operate this way for at least 30 minutes. After 30 minutes we would have operator action in the control room, and that would be this alignment here.

Where, again, we still have the maintenance on this -- on train one. We have train two that is -- has a single failure. We would close -

- we would open the cross-connect valve from train four to feed train one's steam generator, and we would also isolate steam generator four.

So, after 30 minutes we would have the storage full and the pump aligned from train four to feed the steam generator in train one, and we would still have the intact steam generator of three. And that's sufficient for our Chapter 15 analysis.

MEMBER STETKAR: I understand Chapter 15 analysis and meeting all kinds of criteria, but in the real world, can I be -- can I remove post trip decay heat by feeding one and only one steam generator with one and only one emergency feedwater pump?

MR. CONNELL: Jim, we looked at that scenario.

MR. McCUMBER: Initially with that one pump you will not.

MR. CONNELL: You will not?

MR. McCUMBER: The decay heat is higher for the -- I'm sorry. With one pump operating and using --

MEMBER STETKAR: The 400 gpm --

MR. CONNELL: Well, the rated flow is still -- you can't --

MR. McCUMBER: Well, it differs depending on if you have reactor coolant pumps running and things like that. When you don't have pumps running, it's fairly close.

MEMBER STETKAR: It's fairly close, but a plan vanilla turbine trip, let's say, or plain vanilla reactor trip you can't quite -- oh, okay.

MR. CONNELL: But we have four.

MEMBER STETKAR: I understand you have four. That's -- I was just trying to understand what the minimum realistically required. So minimum realistically, if I think -- again, PRA success criteria, you need sort of one plus. You need two.

MR. McCUMBER: No. Let me clarify that, too. Chapter 15, using the ANSI standard for decay heat, you know, if you go to more of a best estimate PRA type of decay heat, one pump can do it.

MEMBER STETKAR: In a best estimate sense, then let me ask you, because you're talking about this cross-tie. How long do I have before I need to open the cross-tie valve to a second suction tank?

For example, if I only had -- if I only had either pump two or three running --

MR. McCUMBER: Six- to eight-hour time frame.

MEMBER STETKAR: -- on a smaller tank?

MR. McCUMBER: This is to get another tank aligned, you're saying?

MEMBER STETKAR: Yes. In other words, suppose I only had one pump in the world, one emergency feedwater pump and I needed realistically to remove decay heat, how long could I do that with the available inventory in its respective storage tank?

And I'll pin you to the small of the two storage tanks. I'm trying to figure out a time window for how much time is available for the operators before they really need to open the cross-tie.

In your example you said, well, I can assume after 30 minutes, as long as I can survive for 30 minutes, according to design basis accidents base I can take credit for the operators opening those valves.

I'm trying to understand how long is really available before they have to open the valves.

MR. McCUMBER: Okay. We are dealing again with the supply cross-connect?

MEMBER STETKAR: The supply cross -- I'm strictly talking about water inventory.

MR. McCUMBER: Okay. With one pump running --

MEMBER STETKAR: I mean, I just didn't --

MR. McCUMBER: If we're pumping 400 gallons a minute --

MEMBER STETKAR: Well, you're pumping whatever you're pumping to maintain steam generator levels, so I'm assuming that the flow control for the level control valve is going to back off on you.

MR. McCUMBER: Yes. Initially we'd be going at 400. That's what the flow control will maintain a system at, and then it will drop off and level goes. So, as a minimum, we would have the 400 into 98,000 gallons, so that will give you a minimum as far as what the -- when the flow --

MEMBER STETKAR: Yes.

MR. McCUMBER: -- breaks off it will be quite a bit -- you know, later on you get more time.

MEMBER STETKAR: Okay.

MR. McCUMBER: I don't have that figure in my head.

MEMBER STETKAR: Okay. I was just curious whether you had it. I've seen, you know, numbers bounced around like six to eight hours and things like that.

MR. McCUMBER: The six to eight is the time that -- wasn't assuming we only had one pump.

MEMBER STETKAR: Right. I -- that's

right. That's what I was trying to pin you again in kind of PRA realistic analysis space, how much less than that period, but more than 30 minutes might be available.

MR. McCUMBER: It's quite a bit.

MEMBER STETKAR: Okay.

MR. McCUMBER: I don't have the number but it's a lot.

MEMBER STETKAR: Thanks. Thanks.

MR. CONNELL: All right. And we also wanted to address the reliability and diversity of our four motor-driven pumps. Of course, with our feedwater supply system, emergency feedwater supply, our objective is to have highly-reliable emergency feedwater with low core damage frequency.

The system design we have achieves this reliability target of 10 CFR 50.34 through the combination of redundancy and diversity.

We felt as though this was superior to a turbine-driven pump scenario because of the low reliability associated with turbine-driven pumps, extra maintenance requirements of that, and then the associated high-energy line you're bringing down into that.

And just my personal experience as a

systems engineer, I just have personal bias. Now, this decision was made before I joined the group, but I whole-heartedly support this decision to not have the turbine-driven pumps.

MEMBER STETKAR: What would your reaction be to small diesel-driven pumps, direct-drive diesel pumps?

MR. CONNELL: That's an option, but then you have to put it into tornado-protected buildings. You have to accommodate that somewhere. You have to exhaust. It just doesn't fit well inside our plant format.

Okay. And to support this idea of the reliability we've done sensitivity studies to compare four motor-driven pumps to two motor-driven and two turbine-driven pumps. We responded recently to an RAI from the staff on this, and I know the staff's going to touch on this later, so I won't steal their thunder too much.

But through the sensitivity studies we evaluated that the overall system comparison, four motor-drive to two turbine- driven and two motor-driven is essentially equivalent.

MEMBER STETKAR: Okay. Since I get to be the duty PRA man here --

MR. CONNELL: This is the softball we're --

MEMBER STETKAR: What is the source of these numbers? Are they from your PRA model? I mean, these are -- they are numbers. They are precise numbers. You've got the three significant figures.

You do have a PRA. Are these numbers derived from the PRA itself?

MR. CONNELL: Yes. Tim, do you want to go into more detail of that?

MR. TESFAYE: Sure. Yes. These are -- these were based on a sensitivity study from our PRA.

MEMBER STETKAR: Okay.

MR. TESFAYE: And there was an RAI response 238, supplement one. We provided some of the background relative to -- we had to come up with a conceptual design for the two turbine-driven pumps.

We made a number of simplifying assumptions to try to understand the basic sensitivity and just see where do we really stand. And these are the results from that study.

MEMBER STETKAR: Okay.

MR. TESFAYE: I guess I'll be happy to answer any specific questions you have about it.

MEMBER BANERJEE: Do you have a number for four turbine-driven pumps?

MR. CONNELL: We did not do that scenario, no.

MR. TESFAYE: No, we did not do four turbine-driven pumps.

MEMBER BANERJEE: Interesting.

MR. CONNELL: Well, the reliability of the turbine-driven pump itself to the motor-driven pump, that reliability --

MEMBER STETKAR: That's an easy -- easy -- I can't do it to three significant figures, but it's a lot worse.

MEMBER BANERJEE: I just wanted to know --

MR. CONNELL: The reliability level is --

MEMBER STETKAR: I guess that's not the question you wanted.

MEMBER BANERJEE: No, no. Just for comparison. I'm not advocating that you do that.

MEMBER STETKAR: You could do it, but it's sort of not a worthwhile thing to do.

MR. CONNELL: It would be worse, yes.

MEMBER BANERJEE: It would be quite a bit --

MEMBER STETKAR: Well, I don't want to dwell too much on the numbers. If you say they're derived from the PRA, I'm assuming eventually we'll get to the PRA and I can ask at that time questions about treatment of common cause failures and how you did that.

But as long as these are consistent with what's in your PRA, that at least satisfies me for now. Thanks.

MR. CONNELL: Okay. So in conclusion, the steam and power conversion system does what it's supposed to do. It removes energy from the RCS and generates electricity in the turbine generator.

It also protects the public by providing safety-related heat removal via the emergency feedwater system and the main steam release.

MEMBER BANERJEE: Well, you definitely have a much large role here for this system for small breaks because we don't have a high-head injection system.

MR. CONNELL: Right.

MEMBER STETKAR: The bigger role is the -- the bigger role is the steam relief for small breaks because, you know, the emergency feedwater has the same role as small breaks.

MEMBER BANERJEE: No. I'm saying the full system --

MEMBER STETKAR: Oh, the whole integrated system. Yes.

MEMBER BANERJEE: Whatever it is.

MEMBER STETKAR: That's correct.

MEMBER BANERJEE: When is Chapter 15 going to be done?

CHAIRMAN POWERS: We have it listed for July. July, yes.

You are done.

MS. SLOAN: We're done.

CHAIRMAN POWERS: Okay. We're going to take a 20-minute break and then we're going to come back and listen to the staff.

(Whereupon, the above-entitled matter went off the record at 9:41 a.m. and resumed at 9:59 a.m.)

CHAIRMAN POWERS: Let's go back into session.

MR. TESFAYE: The staff's presentation will be led by the Chapter 10 project manager, Peter Hearn. Please go ahead.

MR. HEARN: All right.

CHAIRMAN POWERS: And Pete, the ground rules, same for you guys as for them. You've got to give us a little bit of your background and --

MR. HEARN: Okay. Let's see. I graduated

from Brooklyn Polytechnic Institute with a Bachelor's degree in mechanical engineering. I also have a Master's degree in -- from the University of Maryland in mechanical engineering with energy conversion, which is a thermodynamics, heat transfer and fluid dynamics.

Then I went to work for the Naval Ship Systems Command and I spent almost five years in the Propulsion Systems Analysis Group where we analyzed steam plants, diesels, turbine generators.

And I came to work for the AEC before it was in NRC. I've been here 37 years. In that time I've spent over ten years in the auxiliary and Systems Power Conversion Branch which encompasses the present day Balance of Plant plus more.

I spent over ten years in the Containment Systems Analysis Branch and I spent over ten years in the Technical Specification Branch.

With that, I'd like to start the presentation.

CHAIRMAN POWERS: You sound like a man that can cover the subject.

MR. HEARN: The technical staff consisted of five reviewers, Devender Reddy from the Balance of Plant, John Honcharik with the Components Integrity

Branch, Robert Davis, Component Integrity Branch, Angelo Stubbs from the Balance of Plant, and Jeffrey Poehler from the Components Integrity Branch.

During the review there were 75 questions asked and the items status from those questions amounted to 12. The following slides list the brief description of each of the open items, of the 12 open items.

We'll go to the first presentation. And the first presenter is Devender Reddy.

MR. REDDY: Thanks, Pete.

Good morning, Dr. Powers and company and the staff and to the Applicant.

My name is Devender Reddy, I'm the lead technical reviewer for Sections 10.2 Turbines Generator, 10.3, main steam system and 10.4.1, main condenser and associated systems.

Regarding my background, currently I'm working in the Balance of Plant Branch in the NRO. I joined the NRC just about seven years ago and I have been performing technical reviews for power up rates, licensing amendments and license renewals.

Also I worked in the license renewal division for a couple of years as licensing project manager, auditor and team leader.

Regarding my qualifications, my education, I have a B.S., mechanical engineering and a couple of Master's, and the final Master's is from the University of Michigan in the nuclear engineering where my advisor was, Professor Kerr who was once on this committee.

CHAIRMAN POWERS: A hallowed member of the --

MR. REDDY: Actually, I live in Ann Arbor, so next time when I see him I'll tell him.

At the NRC I received qualification certifications as a reactor technical reviewer, license renewal and project manager and auditor and team leader. That's my background briefly.

And coming to today's presentation, the topics I'm going to talk about are, as I said before, turbine generator, mainstream system and main condenser and associated systems.

In these reviews BOP has been the main branch, however there are other branches such as Reactor Systems Branch, Instrumentation and Controls and Component Integrity Branch.

In these reviews, the staff reviewed, that it was a design for these sections of the EPR design. And the reviews were based on NRC regulations and the

SRP guidance as applicable.

In the process of these reviews, the staff issued several RAI's and received proper responses from Applicant. Based on its review of the DC, of the design certification application, the staff found EPR design is acceptable, except there is one open item that is with regard to the turbine overspeed production system.

With respect to the turbine generator system, the DEH-C, of course, you may all know that it's transfer digital electrohydraulic and control system.

MR. TESFAYE: On that point I'd like to mention the fact that we have the acronyms at the back of the slides.

MR. REDDY: I think it is not there.

CHAIRMAN POWERS: Thank you. Very useful.

MR. REDDY: Yes. As I said earlier, the staff's review is based on the NRC regulations such as general design criteria, what we call GDC in short, and also SRP guidance.

With respect to the turbine generator overspeed system, and generator in general, conformance to GDC-4 requires that for the protection of SSE's, that are important safety from the turbine resource, the turbine should be provided with a turbine production system to minimize the probability of no missile generation.

And whereas the SRP stipulates the GDC-4 requirements as to meet the GDC-4 criteria, the SRP acceptance criteria, Section 10.2 specifies that the turbine production system should include an overspeed control system with suitable redundancy and diversity features.

Whether MSRP guidance calls for proving a mechanical overspeed device to protect the turbine at 111 percent of its rate of speed, and emergency backup electrical overspeed system device at 112 percent of the rate of speed to meet the redundancy and diversity factors. The maximum allowable is 120 percent of the rate of speed.

So the staff reviewed the EPR design, whereas EPR design uses two electrical overspeed systems, and install one mechanical and one electrical.

And we, the staff, reviewed the EPR DC, the design certification application, and focused our review on the redundancy and diversity features.

In the process, we issued several RAI's and received proper responses, as I said, for additional information and clarifications we got from the Applicant.

And based on our review and the AREVA response, the staff found the additional information was acceptable because it conforms to the Commission Regulations in the SRP guidance, except that the staff requested for a Tier 1 ITAAC to confirm the various features of the two electrical systems to conform to 10 CFR 52.47(b)(1).

What is the -- I summarize the 10 CFR 52.47(b)(1). It requests that a design certification application to contain an ITAAC, will ensure that when the plant is built and the inspection and testing and analysis are performed, a plant that uses the design certification is built and operated in accordance with the design certification and also NRC regulations, and other applicable Commission rules.

So, for that, based on that we issued an RAI, the question is 10.2-7 in this regard, and this remains as an open item.

MEMBER STETKAR: I learned this morning that the -- although there are diverse and redundant overspeed trip input signals, there, indeed, is only a single trip block that must operate to trip the turbine in the overspeed position such that even if I had 500 redundant diverse overspeed trip input signals of that, if the solenoids for that particular trip block have a problem, the turbine doesn't trip.

Is that at all concern to the staff? Have you thought about that feature of the design?

MR. REDDY: Yes, but trip blocks actually are -- I didn't find those, you know, they discussed over the trip blocks, and we evaluated that and based on our evaluation we found it acceptable.

MEMBER STETKAR: But you used the term, plural, trip blocks.

MR. REDDY: Yes.

MEMBER STETKAR: I used the term trip block, singular. And, as I understood from the discussion this morning, there is a singular one and only one trip block that has three solenoids associated with it.

So, there's not a redundancy in terms of trip blocks. There's a single -- if I can call it an electromechanical device that ports some hydraulic fluid around.

MR. REDDY: Yes, that was -- actually my control support is not here, but let me look at AREVA, if they can answer the question, otherwise, I'll get back to you.

MR. CONNELL: The reliability of that trip lock we feel is very, very high reliability. Were there any other aspects of it, though, Bob, that you can add to that to-- for the redundancy capability?

MEMBER STETKAR: Or lack thereof.

MR. DAY: The thing about older turbine designs is not so much does it have electrical versus mechanical, is that there are separate sets of valves or mechanical devices or electrical devices that dump the hydraulic pressure so that -- so that, indeed, whether it's an electrical trip or an old-style Rube Goldberg mechanical trip, indeed, the hydraulic fluid was released from different points in the system so that there, indeed, was a redundancy all the way out to the point where you dump the fluid.

Here -- here you're relying, I believe, on a single trip block. It's a multi-port trip block. It has three sets of ports in it, but it, indeed, is a single device with solenoids that need to move little plugs in there to position ports whether they are open or closed.

It is a little hard to see how this is diverse --

MEMBER STETKAR: It's really hard. Yes. I mean, --

MR. DAY: It's also hard to see how one of these things work unless you sort of look at it a lot.

MR. CONNELL: Do you want to address that?

MR. DAY: It was a schematic that was part of, I think, the RAI response to 243.

MS. SLOAN: Bob, do you want to identify yourself and use the microphone --

MR. DAY: Oh, Bob Day with AREVA. That would show the way that it goes into a single trip block. Now, the trip block is designed that if it fails, either by loss of power, loss of hydraulic fluid, that it trips the turbine.

MR. WIDMAYER: Can you repeat that RAI number, please.

MR. DAY: I believe it was 243.

MR. REDDY: Do you have the question number, Bob? The question number, do you remember?

MR. DAY: I think it was 10.2-6.

MEMBER STETKAR: There was -- in the SER, actually. The thing that started me on this was, in the SER in Section 10.2-41 of the SER, it mentions FSAR Tier 2 figure 10.2-61, it shows the overspeed trip system schematic and figure 10.2-62 shows the turbine trip block schematic but, indeed, in the FSAR those figures don't exist.

So, I couldn't actually go find something to look at even though the SER says it's in there somewhere. That's why I asked the question about it this morning, because I'd seen the term "trip blocks," plural, and I think we've confirmed that there's only -- at least we've confirmed there's only a single trip block.

MR. REDDY: Yes. We will have to go back and check up on that and then I'll get back to you on that.

MEMBER STETKAR: Okay. Thanks.

MR. REDDY: Okay.

CHAIRMAN POWERS: I would like to get back --

MR. REDDY: Sure.

CHAIRMAN POWERS: -- and know the status on this.

MEMBER STETKAR: 2.6, yes.

MS. PEDERSON: Okay. This is Ronda Pederson. So we just found that we responded to RAI 2.3, Question 10.02-6, and in that text we stated "The trip block provides interface between the electrical and hydraulic systems and consists of three trip solenoid valves."

MR. DAY: Ah. It does show a schematic,

too.

MS. PEDERSON: They have a schematic.

MR. DAY: And it has a schematic system --

MEMBER STETKAR: Ah, there it is.

MS. PEDERSON: We have a hard copy there.

MEMBER STETKAR: Well, that's a pretty simplified schematic, what the thing really looks like, but -- yes.

MS. PEDERSON: Okay.

MEMBER STETKAR: Anyway, the real question is are you satisfied that you meet the redundancy and diversity all the way through the final device that dumps the hydraulic fluid, that reduces the hydraulic fluid pressure so that the turbine actually does trip.

MR. REDDY: And the question, Doctor, actually it is -- we are talking about there is only one trip block.

MEMBER STETKAR: There's a trip -- there is one trip block.

MR. REDDY: One trip block.

MEMBER STETKAR: It has --

MR. REDDY: More redundancy in there, and not --

MEMBER STETKAR: That's correct. I mean, you need two out of three of the solenoids in that trip block to deenergize to trip the turbine. And although there are redundant input signals to each of those solenoids, there's, I believe, only three solenoid valves.

MR. REDDY: So you have located the input signals?

MEMBER STETKAR: Oh, yes, yes. No, that's fine.

MR. REDDY: But the only thing is, there is only one trip block and that may be not adequate.

MEMBER STETKAR: And I am not sure whether that satisfies, you know, the staff's criteria.

MR. REDDY: No, actually, you know, there's the redundancy we looked for very carefully and that was both of them, so yes, we'd like to get back to you on that. Thank you to the good question.

And if you don't have anything on the ITAAC what we are asking, we do not have any other open items with regard to the turbine generator as relates to the mainstream system and main condenser and associated systems.

So, that concludes my presentation, and thank you all very much for your patience and if you have other questions please show me.

CHAIRMAN POWERS: Any other questions on

this issue? I think we've got it, an issue we need to follow up on at some point. Probably -- probably we can enter that in as an item. Can we do it in the February meeting? Is that when you give a sufficient-- we'll consider having it as part of the February meeting.

MR. REDDY: Excuse me, Doctor. Well, I think John wanted a February meeting, yes. Even before that, we'll get that response to you.

CHAIRMAN POWERS: Yes, but we'll have to do it in a -- we have to allow the public to hear about it.

MR. REDDY: Of course. Yes. If I need further clarification I'll contact you on that.

And if we don't have further questions, thank you very much.

CHAIRMAN POWERS: Thank you.

MR. HEARN: That brings us to the second speaker, who is John Honcharik.

MR. HONCHARIK: Good morning. My name is John Honcharik. I'm a senior materials engineer. I've been working here at the NRC for seven years, previously in -- during the operating plants, and now for the past year, two years here at New Reactors.

I have a Bachelor's degree in

metallurgical engineering from Brooklyn Polytech, and my prior experience included 15 years working at Newport News Shipbuilding as a materials engineer, working on the naval reactors for aircraft carriers and submarines, construction and also refueling of the reactors. And that's basically my background.

CHAIRMAN POWERS: Good.

MR. HONCHARIK: I guess today we are going to talk about the -- I want to talk about the turbine rotor integrity. The review determined integrity of the turbine rotor includes the materials, the rotor design and inspections of the turbine rotor.

This review is based on SRP 10.2.3, and GEC-4 to ensure that the turbine rotor uses materials with adequate material properties, including fracture toughness. A proven design is used, and also in-service inspections ensures that the flaws are minimized and do not grow too bad.

Therefore, we can maintain that the integrity of the turbine rotor will have a low probability of generating a turbine missile.

I'll discuss these three issues separately, but I note that these issues collectively ensure the integrity of the rotor. Okay.

First I'll discuss some of the material

issues that are noteworthy. The FSAR states that low-pressure turbines are manufactured from material that have the nearest equivalent material specifications for ASTM, A-471.

Also, the FSAR states that train elements are controlled and flaws will be minimized and have improved toughness.

Now, based on the staff's review in accordance with the guidelines of SRP 10.2.3, there are two open items associated with the materials for the turbine rotor, in that the specific material specification or additional procurement requirements such as mechanical properties should be specified in FSAR so that the rotors are procured to a specification that will ensure the integrity of the rotor.

Also, a description of the proven procedures used should be specified that will be used to manufacture these rotors to ensure that they have enhanced properties so that they can minimize flaws and prove toughness and minimize chemical segregation thereby mitigating the potential of ruptured rotor.

And finally, the FSAR should include a method of calculating the fracture toughness of the actual turbine rotor material.

Next I'll discuss the design of the turbine rotor. Yes.

MEMBER SHACK: These things are all good, but still, until I actually have the fracture toughness I can't really go through that analysis I need to go through.

So when you say this ensures it, it helps ensure it --

MR. HONCHARIK: It helps ensure. Right.

MEMBER SHACK: But we're still a long way from ten to the minus four probability just knowing this.

MR. HONCHARIK: Right. And also the -- well, the analysis wasn't included in this section. It's included in 3.5.13, turbine missiles.

MEMBER SHACK: Right.

MR. HONCHARIK: And in there they'll have the COL applicant provide that analysis and they have, and in there, you know, what I really want to do is make sure that what they are specifying here in -- for curing this rotor will be bounded by that turbine missile analysis.

Next, concerning the design of the turbine rotor, the FSAR wasn't really clear on whether the rotor is forged or welded, and also the FSAR stated that the inspections and bores is required, applicable.

Now, based on our view there are two open items associated with the design, in that the FSAR should provide, I guess, a clarification of the rotor design using forgings that are welded together based on RAI's responses that were received.

Also, the location of these welds should be specified and AREVA should discuss the inspectability of these welds, especially for the nonbored rotor.

The staff also requested operating experience of these nonbored rotors and the discussion on accessibility of inspecting the rotors and the reliability of the inspection results.

And finally, the staff requested information on how the mature properties of the internal region of nonbored rotors are not degraded due to solidification and forging of such a large item. It should be noted that past versions have removed this internal region.

And now I will discuss some of the issues concerning in-service inspection. The FSAR stated that the COL Applicant will provide the inspection, internal. The FSAR also stated that visual and surface exams of the turbine rotor should be performed.

And finally, the FSAR provides an ITAAC which specifies that the turbine rotor integrity analysis should be performed using the turbine design and material.

Based on our review there are three open items in this area. One is the COL applicant should provide an inspection program in addition to an inspection interval. Also, the FSAR should include a volumetric inspection such as ultrasonic examinations of the rotors which meets the guidelines of SRP 10.2.3, and which is currently industry practice.

Also, need to clarify the ITAAC as specified, that the turbine rotor integrity analysis will be performed for the as-built rotor with the material properties of the as-built.

All these issues are being worked with AREVA, and there is a path forward for these issues, so even though there are, you know, a few open items here, that most of these are resolvable just with especially, you know, most importantly to provide the operating experience of these nonbored rotors and if there's any other issues for those, including inspections of them.

And that concludes my talk on the turbine rotor integrity.

CHAIRMAN POWERS: Go on.

MR. HEARN: Our next speaker will be Bob Davis.

MR. DAVIS: My name is Bob Davis. I'm a senior materials engineer in the Component Integrity Branch in NRO. I've been with the NRC for almost seven years prior to working in NRO.

I worked in NRR on ND welding and materials issues in the Division of Integrity, Division of Component Integrity, DCI.

I have a Bachelor of Science degree in welding engineering from the Ohio State University. Prior to coming to the NRC I was a senior welding engineer with Constellation Energy.

Prior to my professional career I was a welder for 13 years, six of which was in the Navy Nuclear Program.

Today I'll be presenting Section 10.3.6 on steam and feedwater system materials.

The materials used for class 2 and 3 components meet the ASME Code Section 3 requirements, and those materials specifications and grades are listed in the FSAR.

The main steam and feedwater system fabrication follows applicable regulatory guides, such as Reg Guide 1.37, 1.50 and 1.71. And the main steam system and main feedwater system materials meet the applicable fracture toughness requirements of the ASME Code.

The first thing I want to say about FAC, I know that came up earlier in AREVA's presentation, is that there's two parts to this. One part is to design to mitigate FAC, and then the other part would be the FAC program or the appropriate COL item for an FAC program.

The EPR design incorporates design features to mitigate flow accelerated corrosion in main steam system, main feed water system, condensate system, steam generator, nonsafety-related power conversion systems.

Typically this section deals with class 2 and 3 components, but we always ask questions about the nonsafety systems, which is, we all know, are those are the systems that usually have failures and perfidy with the nonsafety systems.

The EPR design features to prevent FAC include material selection, limits on flow velocity and water chemistry -- oh, okay. We missed one. The material selection, limits on flow velocity and water chemistry, in the end, at the final design, AREVA will perform an analysis to determine exactly which parts or portions of systems are susceptible to FAC, and in the class 2 and 3 components, those systems -- those components will be fabricated with a material that has at least .1 percent chromium.

That comes from the most recent version of the EPRI Guidelines on the FAC program which identifies materials with at least 1.0 percent chromium as being FAC-resistant after they are inspected one time.

So, if they use the .1 percent chromium, after the -- which they will have to -- they'll do preservice inspections to establish a baseline, and then after the plant goes into operation, the COL applicant will perform an inspection to verify that that is true, that those materials are not susceptible to FAC in the environment that they're in.

MEMBER SHACK: Now, do you have a higher requirement for FAC resistance because they are going to depend on leak before break and we want materials that essentially have no known failure mechanisms?

MR. DAVIS: I don't think leak before break, does that apply to class 2 and 3? It may class

1.

MEMBER SHACK: It applies to the steam line in the valve room, as I understand it, if they will --

MR. DAVIS: I think typically the main steam lines aren't quite as susceptible in FAC.

MEMBER SHACK: They aren't quite as susceptible, but the question is, is it good enough. You know, what's your criterion -- you know, I'm trying to think back to what the standard review plan or the branch technical position says for leak before break and it says, you know, it has no known active mechanisms.

Well, does carbon steel with .1 chrome in a dry steam -- you know, it's a little iffy, it seems to me, you know, to say it's immune. It's certainly resistant and it's not particularly susceptible, but I'm just curious from the leak before break argument, you know.

I agree with -- that they weren't depending on leak before break, I wouldn't have a problem, and it is an interpretation of the branch technical position as to whether it's resistant enough in a leak before break sense.

MR. DAVIS: I'm saying if it is dry steam, it's probably not susceptible to FAC, even if it's made out of plain carbon steel.

MEMBER SHACK: It's not but, you know, is it immune?

MR. DAVIS: I think that within the EPRI guidelines they -- you know, they have a program that meets the upper guidelines which is -- has evolved from, you know, from the beginning.

We just started off with 1394, whatever the EPRI document, Newmark and all those things, and has progressed up to Revision 3, which includes all the lessons learned and what the industry knows to date.

But as far as how that affects leak before break, probably have to ask that question when the leak before break comes before there. I don't have an answer for that.

And again, like I mentioned before, because they're using .1 chromium, the guidelines still require them to verify that there's no degradation.

Of susceptible nonsafety-related systems may use chrome moly or stainless steel but of course they would be much more susceptible to FAC than the class 2 and 3 systems.

There's a COL information item that requires COL applicants to develop and implement prior to initial fuel loading and FAC monitoring program that conforms to Generic Letter 89-08 and EPRI -- EPRI tactical report for recommendations for an effective flow accelerated corrosion program which is acceptable enough.

And keep in mind that the Generic Letter just says they have to manage FAC, not really exactly what they have to plead the EPRI report is acceptable to the staff, because it is very comprehensive and relies on several years of operating experience.

We have two open items, the applicant did not specify weld filler material classifications and specifications was expected, just like piping or other components, you have to list the specification and the grade, and they did not include the grade in the materials or classifications for weld filler materials was referred to.

The other open item is related to -- which kind of falls back to the .1 chrome issue. The ASME Code requires that degradation mechanisms be taken into account during the design so that the system will maintain its minimum wall thickness through the design life of that system, whether it's the design life of the plant or less than that or, you know.

So, in response to one of our RAI's which we requested that they maintain that they would meet the 60-year life of the plant, the FSAR was changed to say that they would design piping so that it would meet a 40-year design life, and we have an open RAI that, if the design life of the plant is 60 years, then the piping should be designed to 60 years and not 40.

So, although they -- we licensed the plant for 40 years, so they could design it for 40 years but they need to specifically state that it is 40 years, and that that's different than the other 60 or statements that they make.

Do you have any questions?

(No response.)

MR. DAVIS: Thank you.

MR. HEARN: Thank you.

Our next speaker will be Angelo Stubbs.

MR. STUBBS: Okay. Good morning. My name is Angelo Stubbs and I'm feedback systems engineer in the Balance of Plant Branch.

I've been with the NRC for about seven years, originally in NRR, and then once we got into new rad design, the division was created, NRO. In both cases, in Balance of Plant Branch.

Prior to working at NRC, I worked for Stone & Webster Engineering Corporation in the Cherry Hill, New Jersey office. There I was in the mechanical division in a number of different groups, including the radiation protection, the mechanical, heat balance and engineer safety systems and analysis group and for radiation protection engineering, safety systems and analysis I've been a lead engineer and a group supervisor.

I have a background -- my educational background is in nuclear engineer. I have a degree in nuclear engineering from the University of Florida and I've take about 30 hours of graduate -- completed about 30 hours of graduate course work in nuclear engineering at North Carolina State University.

I'm the lead technical reviewer for DCD, Section 10.4.9, emergency feedwater, and Section 10.4.9 of our ISCR will be summarized, the evaluation that would be performed on an emergency feedwater system and their associated storage tanks.

And in that section of the SER we identify two open items associated with that review, and those open items are what I'm going to be talking about today.

The first open item, as we have -- the first open item I'd like to discuss is the open item that we issued because we found conflicting information in the FSAR.

I'll give you a little background. This open item came up because, in response to one of our RAI's there was a valve decision that was normally open, that was changed to normally closed.

And in their RAI they marked up one section and we didn't find that. The other section was marked up. We did review RAI and we were okay with their response and we felt that the issue was resolved, but the failure to get change order sections in the FSAR, we sought that in the RAI.

As you can see in the slide, the issue was associated with Sections 5.4.7.3.3 and 10.4.9.3. In 5.4.7.3, they identified operator action outside of the control room as possibly being necessary under certain conditions, and in 10.4.9.3.3 they did not identify the possibility of having to have operator action.

So, this came out of our review using SRP 10.4.9, and the requirements -- or not requirements, the guidance in Branch Technical Position 5-4.

And as I indicated, and we have it in the slides, this was a matter of resolving the inconsistency. And because we have reviewed the RAI, we felt that their response to the RAI, that -- and the use of operator action was -- was acceptable. Go to the next slide.

Was acceptable. So, what were referring to was using operator action six to eight hours into the event if you didn't have -- if you had a -- if you needed to use a different storage tank and you had train out for maintenance or a break -- a line breaking a train where you needed to use the inventory in one of those tanks.

That's allowed as Branch Technical Position 5-4 for limited operation -- operator action outside the control room, and the six to eight hours is sufficient time to allow that to happen.

MEMBER STETKAR: They also mentioned -- didn't they increase the minimum required capacity of the storage tanks a bit also?

MR. STUBBS: Okay. That was -- I'm not sure whether that was more or whether it was a different one but, yes, the storage capacity or inventory required to get the cold shut-down was increased from about 300,000 to 360,000 gallons, and that was in response to, I think, a different RAI.

MEMBER STETKAR: Oh, okay.

MR. STUBBS: Jason is going to talk about that, about that a little bit. Now, the number that we have now is pretty consistent. It's very close to what we have for the US APWR. I think it's 372,000 gallons, and if you look at the large DWR's, they are about 300,000, but they are about 20 percent lower in thermal power.

So, this all -- you know, this is -- this was all pretty much consistent with what we felt we should have been at, and we did accept that also in our SER.

MEMBER STETKAR: The only reason I asked that is I came back to the question I asked earlier regarding realistic analysis versus design bases analysis and the justification that you have for design -- in design basis accident space that you have about six to eight hours before the operators would need to do something.

I mean, that's all tied in with the amount of water that you have and what assumptions you have --

MR. STUBBS: Right. If you're -- yes. How much is in their tanks. They're slightly -- the two tanks are a little bit larger than the other two tanks but, yes, the total inventory available is soft hydrogen gas.

MEMBER STETKAR: Thanks.

MR. STUBBS: Okay. If we move -- and as far as that, the status of that is -- we brought that up at a previous public meeting and -- and I think AREVA recognized this as just something that needs to be resolved, and we don't think that we'll have a problem closing that out.

Okay. We'll go on to the next. Okay. The second item has to do with, you know, EFW diversity. As indicated, the issue came up because our guidance speaks to diversity, and really the -- if you look at the current US plants and you look at the other active PWR designs that have either been certified, like System 80-plus or is under certification, like US APWR, all of these have the first EFW systems.

And the EPR has only motor-driven pumps as they described to you earlier. And our guidance wants us -- has us look at diversity, and the branch at position 10-1 talks about AFWS pump drive diversity and power supply diversity.

The second item on the regulatory basis which is now what's the requirements, the diversity isn't really in them, but by having diversity, it helps you meet those -- the regulatory bases.

And I listed three -- three of the highlights that we looked at when we are looking at evaluating this system. The first one has to do with the design capability and reliability, really the reliability is the key there, and that's the 10 CFR 50-34.

The general design criteria 34 and 44 deals with redundancy of components, and since CFR 50.63 is a station blackout rule, and in this case we have -- we don't have a turbine-driven pump. We only have motor-driven pumps.

If we get into a situation where we lose that off-site power and lose all the emergency diesels, then we're going to have to do something to get the station blackout diesels available, and there's going to be some time that we're going to have to go without having power available for the emergency feedwater pumps, since they are all motor-driven.

Okay. If we could go to the next slide. On this slide what I wanted to try to do is highlight -- even though that we don't have the diverse -- the diversity in the pump drives, I wanted to highlight some of the design configuration, some of the key design features that we have, that are -- that is in the EPR that really helps us meet those reliability and redundancy goals that we have.

The first is that in terms of the electric power, there's four safety-related class 1 emergency diesel generators and there's also two station blackout diesel generators and those generators are in trains 1 and 4.

And this compares to what we have at plants now as maybe two diesel generators, so you really have four large diesels and six overall diesels available.

And for the -- as far as the system's concerned, the emergency feedwater system, as they showed you earlier, is set up so that any train can be fed by any other EFW pumps, and any pump can be fed by -- can be supplied by any of the storage pools.

So, you have the ability to work with the equipment that you have to provide the steam generators with the water needed to remove the decay heat, whether you have a train out or whether you have a break that they showed to you earlier.

The technical specifications for this plant pretty assures that you'll have a minimum of two -- and in most cases you'll have three diesel generators available for your emergency diesel generators.

When you get to a point where -- where you have two diesel generators available, you'll go into LCO and have 72 hours to return one to service, as compared to -- you get to that point now with the plants out there, when you're down to one diesel generator.

So, this -- not only do they have more -- the four diesel generators, they're going to be available more often. You're not going to be in a situation where you're down to your points with diesel generator.

And the last point on this slide, on this page is the -- they meet the reliability target that they have in SRP 10.4.9. I think it comes somewhere in 10 CFR 50 -- I don't have the number for that.

And that -- again, that was one of the goals, I think, for the diversity, was to get them to meet the reliability target.

MEMBER STETKAR: Angelo, I have to admit complete ignorance of that. What is that reliability target, in terms of numbers? I didn't have a chance to look it up and I just --

MR. STUBBS: It's ten to the minus -- I

don't want to -- if somebody over there -- I don't have it right with me. Is it ten to the --

MR. CONNELL: Between a range of ten to a minus four, ten to a minus five.

MR. STUBBS: And, you know, the SRP is still -- is specified as ten to a minus four to ten to a minus five. They give a range.

MEMBER STETKAR: How much credit is the staff taking for the fact that the current computed reliability meets this target? I know you still have an open item, so I don't want to steal any of your thunder, I mean, but how heavily do you rely on this numerical estimate of reliability versus different aspects of demonstrating diversity and redundancy?

MR. STUBBS: I mean, this -- this is just one factor and this is one factor that's -- that's applied in the SRP, and as -- we'll go on and there's other things that we looked at.

MEMBER STETKAR: Okay. I'll let you continue.

MR. STUBBS: This is an excellent story slide. I was just trying to point out some of the things in the design, because this design was not the same as what we've seen in current plants and the other plants.

The other thing that we wanted to look at, like I said earlier, is a station blackout because in a station blackout we don't have the turbine-driven pump or diesel-driven pump and that's going to take some time after we get into the event before we're going to able to deliver feedwater to the steam generators.

So -- but in that event we're still going to rely on the -- we still have to rely on the station blackout diesel generators and in Chapter 8 they address the availability of the steam generators and the loading and what's -- what they are capable of doing.

But one of the things that station blackout rule does is requires that there's diversity between the station blackout diesel generators and emergency diesel generators.

So, they're -- mechanically that's physically different and separated so that they are not susceptible or they're not as acceptable to the common mode or common cause failure because in a case of a fire they're in different areas.

In the case of components that -- they are different sizes. I don't know if they are different manufacturers, but they have different components associated with it in terms of air start or -- I mean, being air cooled and water cooled.

So, they're not -- that is something that helps the reliability in terms of when -- if we need the stations, I've got these and they will be available.

So, what we did was, we wanted -- we looked at all of that and then as part of our evaluation we wanted to take a look at -- take a look at some things.

And one of the questions was: If we -- you know, not do we just do we meet this, but how does this compare to if just there was diversity there? And we asked the applicant to take a look at that and one of the things that they did was, they looked at an alternate design that used two turbine-driven pumps to see what impact it would have on the PRA.

And what they found out was that in the integrated PRA perspectives, that the first pumps will not be expected to reduce risks significantly.

So, that was one thing that -- that we wanted to look at in a comparison. We didn't look at the -- only turbine-driven pumps because we are looking at diversity -- the impact of diversity.

So, that turned out that we -- it didn't seem like going to an alternate design just to incorporate diversity was buying us a whole lot or was buying us anything.

MEMBER STETKAR: Let me ask you a little bit about that.

MR. STUBBS: Okay.

MEMBER STETKAR: And I guess I'll ask AREVA. This is not a risk-informed license application, is it?

MR. CONNELL: No

MR. CONNELL: So we can't really rely on numerical risk information to make a final decision, but it's pretty clear to me as a risk person, that the design with two turbine-driven pumps versus -- and two motor-driven versus four motor-driven wouldn't necessarily show much if any benefit.

Other combinations of three and one with different types of drivers might. So, just simply so that being a particular design that on the surface doesn't look like it would be more reliable in showing that it isn't doesn't necessarily answer the question would other -- other types of diverse designs actually be more reliable.

MR. STUBBS: That's true. And I understand what you're saying --

MEMBER STETKAR: Turbine-driven pumps are not very good, I agree. Just saying I have a diverse turbine-driven pump is not necessarily the best thing to do in the real world.

MR. STUBBS: Right. I understand, and I guess my concern -- one of my concerns if it has to be one, you know, that if you have something out for maintenance, you know, if you'd put it back into a situation that you don't have diversity again.

MR. HEARN: In 1950 the American Railroads had a similar situation when they converted from steam to diesel electric. It's a very similar situation. A diesel electric locomotive has a diesel driving a generator which supplies power to motors that turn the wheel. Their system turns an impeller on a pump.

And they eliminated the common failure concern by testing inspections and design features. And you know it worked because the American Railroad system has nodes where only one train can go through at a time and they have single tracks.

If they ever had a problem with common failures in this type of propulsion system we would have severe shortages in things we use every day. So, just resign on the side of this design.

I may give you other examples, too, of

that.

MEMBER STETKAR: Thanks. I just have to be careful again -- I want to be a little bit careful about relying too much on numbers that are associated with predefined assumptions about what a design needs to look like.

They can be helpful, but in some cases without examining a variety of different options you might not necessarily have that -- that spectrum of reliability values.

MR. STUBBS: I understand. And one of our major -- our major focuses is we're relying on having the power there and having the four emergency diesel generators, what's the tech spec for those diesel generators, what's the availability of a station blackout diesel generators and what they are powering.

They are powering one in four, it would just be the one. So, -- but in addition, we wanted to understand just because we could be successful with that path, we want to understand whether there was -- whether we were losing ground on what we had already done. So, we wanted some type of comparison there.

The next thing that we really wanted to take a look at was a station black-out event and, as I said, in Chapter 8, they outlined that event. We can go to the next one.

Basically the station blackout diesels are started within 10 minutes for the state of the emergency feedwater system, we're able to deliver feedwater to the steam generators in 30 minutes.

So, what we did was, we went to AREVA and we wanted to know how long will it be before we get to the situation where we're approaching steam generator dry-out and we asked that in our RAI, and their response was one and a half hours.

And that far exceeds 30 minutes that it's going to take to deliver the emergency feedwater to the steam generators. In that response we didn't really -- we weren't able to see what they were, how they came about that number, so we followed that up with an audit of their evaluation so that we could see what we're into and what assumptions and how they came about that number.

And we had Reactor Systems to take part of that audit and afterwards we did a confirming -- we did an independent evaluation and our independent evaluation came up with a number similar to the one and a half hours.

So, we felt comfortable that, in a station blackout, that using the station blackout diesels and having the flow available at 30 minutes, there was enough thermal capacity in the steam generators so that that wasn't a problem.

MEMBER STETKAR: I don't remember because it's all of probably two and a half weeks ago or less, and I'm lucky I can remember this morning.

Can they start and load the station blackout diesels from the main control room in this plant, or is a local action out in the -- wherever the diesels live?

MR. STUBBS: We have AREVA.

MEMBER STETKAR: I should know this because we looked at the hydraulic system two weeks ago. But, as I said, I --

MR. GARDES: I am Brian Gardes. I did my bio when I was here a couple of weeks ago for electrical, so hopefully that will suffice.

The station blackout diesels will automatically start on a loss of off-site power events. They are connected into the nonsafety buses. The operators in the control room will close into manual breakers and energize the class 1 safety-related buses.

At that point they'll be available to restore HVAC, emergency feedwater as the procedures dictate.

MEMBER STETKAR: Thanks, Brian. The neurons fired. Thanks.

MR. STUBBS: The emergency feedwater, actually, I think the start-up will be on low water level on the steam generator, so it may be sometime after 30 minutes before they even get a start.

MEMBER STETKAR: Yes. I was just more curious about getting power to the pumps from the SBO diesels, and I forgot the arrangement. Thanks. Okay.

MR. STUBBS: Okay. So --

MR. TESFAYE: Excuse me, Angelo.

MR. STUBBS: Yes.

MR. TESFAYE: This is Tim Stack from AREVA. Can we -- when we've talked about the diversity question, I guess we'd like to make sure that we understand the values, the quantitative values we've reported are based on secondary heat removal only.

We have safety grade coolant in the supplement that diversely, for core cooling.

MR. STUBBS: Right.

MR. TESFAYE: That is not reflected in the quantitative results we've provided. So, in the context of did we just meet a quantitative value versus do we actually have diverse means to remove decay heat, the answer is yes, we do. So, it's more just --

MR. STUBBS: And that's true, but diverse means the feed and bleed is still dependent on having electric power available. So, the diversity thing -- well, there is -- you don't have the motor-driven pumps.

The feed and bleed is something available, but because you need a pump to run it, you need electric power, but we are aware of the fact that that is something that you have.

So, I guess, to conclude, what we looked at is, we looked at whether we had the reliability, whether we have redundancy and can we cope with station blackout, and our evaluation we feel like the design does that, but right now we've had supplemental RAI's to try to address all the diversity, and we recently received the responses from that, and right now we're going to evaluating that and depending on our evaluation, if everything is answered, then this can become a confirmatory item and we will -- and actually they updated the effort we are at this stage, and can close.

MEMBER STETKAR: Angelo, if I can ask you, and I know you're still doing the evaluation.

MR. STUBBS: Yes.

MEMBER STETKAR: So it might be a bit premature, but are your -- I'll be careful in wording here. Are your remaining concerns with the diversity of the power supplies to the pumps, or the fact that they just have four nominally identical motor-driven pumps?

MR. STUBBS: Well --

MEMBER STETKAR: Regardless of whether I add a thousand different fully-diverse power supplies.

MR. STUBBS: No. Our remaining concerns is basically -- we've discussed things and we've gotten some partial responses back. It's getting everything back in terms of their responses and updating the actual -- the licensing document, and making sure that everything is captured.

That's our primary concerns. And to see that is -- it matches what we expect to get back. The reliability of the motor-driven pumps, I mean, we have diversity in terms of the power supply. We have the capability of feeding and supplying our pump.

I don't -- I guess your question is just the reliability --

MEMBER STETKAR: No, I didn't -- I was

careful not to say reliability.

MR. STUBBS: Okay.

MEMBER STETKAR: But I was concerned that -- that what I'm hearing you say now is that you -- it seems like you are reasonably confident that this design with the diverse diesel generator power supplies meets all of the criteria.

MR. STUBBS: Yes. With the diverse power supply and the additional redundancy, we --

MEMBER STETKAR: So you're not particularly focused on the fact that the design happens to have four nominally identical motor-driven pumps, that's not the concern about diversity, it's more of a --

MR. STUBBS: Well, that's where we started, and then we had seen enough about the design to feel confident that the design allows us to address the regulatory bases, and that's where we ended up.

MEMBER STETKAR: Okay. And that reliability number is part of something that gives you confidence that you're still meeting that regulatory bases?

MR. STUBBS: Right. If they came up and it was marginal or if it didn't meet that, that would be a red flag for us.

MEMBER STETKAR: Have you looked at all at the analyses that went into support those reliability numbers?

MR. STUBBS: We've had -- well, I don't know if -- we've consulted our PRA Branch in working with this --

MEMBER STETKAR: I mean, I -- we obviously haven't seen the analyses at all and I wasn't particularly concerned about it because eventually we'll dig into the PRA and --

MR. STUBBS: Well, I'll let Teresa --

MEMBER STETKAR: -- circle back to this thing from there, but if the staff is relying -- I don't want to use the term "heavily," but using that information to kind of support your confidence --

MR. STUBBS: Well, that just -- I mean, even if we didn't have a -- the question about diversity, we would still be looking at that number --

MEMBER STETKAR: Sure.

MR. STUBBS: -- because that's the number that we look at when we do our evaluation.

MS. CLARK: This is Theresa Clark from the PRA staff. I haven't given my background. I've worked for the NRC for coming up on about six years, most of that in PRA, in mostly Level 1 and shut-down period analyses.

Before that I got a Bachelor's and Master's in materials science and engineering from the University of Maryland.

We've been involved since almost the beginning of the --

MEMBER STETKAR: Hold on a second. I have to say this for the record. Bill Shack, there's still hope for you.

MS. CLARK: We've been involved almost all along in evaluating this issue with the Balance of Plant Branch, and they brought us in early for risk insights, and from the beginning we really couldn't give them much because a lot of their requirements are deterministic.

They do have that reliability goal, but that's only one part of what they look at. And so, we did some back of the envelope calculations and showed them that maybe on a pure system reliability aspect there could be some improvement from adding diverse pumps, however the way the PRA is constructed and the way the plant's designed there's going to be support system dependencies no matter what.

So, from an integrative perspective we didn't expect there to be a big benefit and that's what AREVA's sensitivity study bore out. So, it's sort of natural. We expected, and all along we've encouraged them not -- as you're saying, not to look at the numbers too much.

And if you want to talk numbers, we'll be back.

MEMBER STETKAR: We will talk numbers eventually. Thanks. I'm glad you did get involved and have thought about that a lot. MS. CLARK: Yes.

MEMBER STETKAR: Thanks.

MR. STUBBS: Okay. So the statuses were the evaluating it and hopefully we'll be able to resolve this in the near term. That's my presentation. Do you have any other questions?

CHAIRMAN POWERS: Any other questions on this subject? You're done.

MR. TESFAYE: Pete, we have some accompanying remarks.

MR. HEARN: No, we have one more speaker. No, we don't.

CHAIRMAN POWERS: Who's going to explain the acronyms.

MEMBER SHACK: I notice RAI is just request for additional, more and more and more.

MR. HEARN: During the review we had several conference calls, audit. We had a public meeting to identify the different open items and the clear pathway to diversity issues within the scheduled resources.

On closing the 12 issues, this chapter should meet the design requirements and regulations.

MR. TESFAYE: Thank you, Pete. I think that concludes the staff's presentation on Chapter 10, SER with open items, and this -- anybody's got questions for us?

CHAIRMAN POWERS: Are there any other questions on the staff presentation in particular?

(No response.)

CHAIRMAN POWERS: None. I'm not going to ask if you're happy. You're never happy.

MEMBER STETKAR: If I was happy it would be an earth-shattering experience.

CHAIRMAN POWERS: Yes.

MEMBER STETKAR: But that's a different issue.

CHAIRMAN POWERS: It would be too hard on your heart.

Okay. On this overall Chapter 10 we did come up with a question that you're coming back to and you'll tell us an answer to this.

It seems to me that we are running into an issue that might be useful for the Committee to explore. It has nothing to do with AREVA. You guys are not responsible for this one.

It is what -- what we mean when we day diversity, and what is diverse and what is diverse enough and what kind of things the staff's looking for when they look for diversity.

And it may, in the end, be completely solvable to the PRA kinds of analysis, but I don't know that for a fact, and we might want to ask the Committee as a whole if they want to explore that a little bit.

I think it depends -- their interest is going to have been that they are running into this issue on other contexts. We run into it enough here that I'm not sure, coming in this morning, Sandra, I thought I knew what diversity was. Coming out, I'm not sure that I know what diversity -- it may be like "obscenity," that I know it when I see it, but in the abstract it may be difficult.

So we may, in our -- when Mr. Bonaca asked me what we did to earn our salary for this month, I can raise that issue with him.

MEMBER STETKAR: I think that the issue has come up -- this is just one.

CHAIRMAN POWERS: Yes.

MEMBER STETKAR: This emergency feedwater, but the issue has come up with some of the other designs in terms of claiming diversity for different contexts.

CHAIRMAN POWERS: I mean, one would like a really crisp definition. And I can see how a crisp definition might come out of a --

MEMBER STETKAR: Well, it's a gray area in the risk assessment business, and once you get to equipment that is, to use the nondescript term "similar," when are the similarities enough, for example, that you would consider it to be susceptible to the same type of common mode or common cause failure, when do you consider that equipment to be different enough.

And I'm not going to use the word "diverse." Different enough.

CHAIRMAN POWERS: Yes.

MEMBER STETKAR: Well, you would essentially say they are not susceptible to the same type of common mode or common cause failure. And it is. It's a gray area. There's not any clear guidance one way or the other.

CHAIRMAN POWERS: It might be useful for the Committee as a whole to at least get an understanding of the grayness, if not a resolution to the issue.

MEMBER STETKAR: Right.

CHAIRMAN POWERS: So, we may want to raise that as something for the Committee to propose --

MEMBER STETKAR: Yes.

CHAIRMAN POWERS: -- unless you just happen to have in your pocket an answer to this question. We would certainly welcome that but, I mean, it does not have to do with your application. It's more of a generic issue. It doesn't have to do with you guys, either, except unless you happen to have in your pocket a very crisp answer to this question.

But it's something that maybe the Committee wants to pursue a little bit to see if they can be as confused as I am on what exactly is diverse enough.

My intention is to -- I think I'm obligated to shift radiation protection to after lunch because they're scheduling public involvement of this.

And so, we're going to have a very liberal lunch break. This will facilitate Dr. Sanjoy's predilections for pheasant under glass and the like for lunch.

MEMBER BANERJEE: AP 1000.

CHAIRMAN POWERS: But, there are no other questions on this particular issue. We are not done with this issue. We will come back to this particular issue, but we've had a nice introduction.

I thank both the people from AREVA and from the staff. I like, by the way, the way you organized the staff presentations and the questions. I thought that was an efficient way to do things.

And so we will recess until one o'clock. That is long enough for the pheasant under glass, right?

(Whereupon, the above-entitled matter went off the record at 11:13 a.m. and resumed at 1:02 p.m.)

CHAIRMAN POWERS: We note that Sandra has abandoned us for Texas A&M and so any task, especially if it's obnoxious and what not that can be assigned to Sandra, we will do so.

Ronda, you're up.

MS. PEDERSON: Okay. Good afternoon. A quick introduction. I'm Ronda Pederson, and I'm the licensing manager for the US EPR, Design Certification for AREVA, and I come with about 20 years of experience. My degree is from the University of Wisconsin, Madison.

Nuclear engineering, and been with AREVA for about four years. Vermont Yankee. I was in licensing, worked on the extended power uprates, 20 percent. Prior to that I was in safety analysis at Point Beach, Accident Analysis and Safety Analysis Group, and then prior to that as a reactor engineer at Dresden Station in Quad Cities and came out of school with seven years for the State of Wisconsin in radiation protection, emergency planning and dose assessment.

CHAIRMAN POWERS: Oh, okay.

MS. PEDERSON: So, with that, I'm pleased to introduce Pedro Perez, one of my colleagues who is a supervisor in engineering in the flow safety analysis.

MR. PEREZ: Good afternoon.

CHAIRMAN POWERS: Good afternoon.

MR. PEREZ: My name is Pedro Perez from AREVA. I have led for AREVA all the radiological analyses in the FSAR, ranging from this one, Chapter 12, Radiation Protection, Chapter 11, the Rad Waste System and radiological effect. And, in Chapter 15, design basis accidents.

I have a B.S. and M.S. degree in nuclear engineering, all of it in basically in supporting safety analyses, half of those in radiological engineering.

What I'd like to do today is present the work we have done in Chapter 12, Radiation Protection. Chapter 12, our work followed the standard review plan and also reviewed anything dealing with interim staff guidance related to Chapter 12, any regulatory issue summaries related to Chapter 12.

We incorporated both applicable, trying to be following the standard review plan as closely to as possible.

The structure of Chapter 12 starts with 12.1, which is ensuring the occupational radiation exposure are as low as reasonably achievable. 12.2, covers radiation sources. 12.3, radiation protection design features. 12.4, those assessments, and 12.5, operation radiation protection program.

In 12.1, ensuring the occupation doses are ALARA, the EPR design really started looking at the worldwide operating experience, looking at where did occupational dose come from and within the design, look at improving those areas, those activities to reduce -- to reduce exposure.

So we looked at maintenance operations, in-service inspections, refueling operations, radioactive waste handling, abnormal plant operations and finally, decommissioning activities.

The US EPR design reflects on operating experience and implements the ALARA principles in the design process. We looked at physical plant layout that includes compartmentalization and dedicated ventilation systems.

This takes into account segregating, physically segregating radiological areas from clean areas to minimize the spread of contamination and also to minimize exposure as people gather to the plant for work.

MEMBER RYAN: Just a quick question.

MR. PEREZ: Yes.

MEMBER RYAN: Undertake examination for segregated ventilation, zone-by-zone, that kind of thing?

MR. PEREZ: Exactly.

MEMBER RYAN: Okay.

MR. PEREZ: The only thing that would share would be the exhaust plenum out to the stack, but there will be --

MEMBER RYAN: So for back-flow, you're okay.

MR. PEREZ: Exactly.

MEMBER RYAN: Yes.

MR. PEREZ: Exactly. The design features prevent, again, the back flow from going --

MEMBER RYAN: And you'll talk a little bit about those details.

MR. PEREZ: Yes.

MEMBER RYAN: Okay. Thank you.

MR. PEREZ: We also looked at material selection to reduce activation corrosion products, and the staff had quite a requests for additional information regarding what materials selections we had, you know, the reason for the material selections and what were we doing to address some operating experience in the US.

So we've looked at using lower cobalt content steels, alloy 690 for steam generator tubes, reducing antimony to the extent possible for reactor coolant pump seals.

I mention that by having this compartmentalized design, you are by the full building permanent shield structures, so there's less necessary temporary shielding during any outage activities.

If you look at the FSAR, you will notice, as you look at plan views, quite a considerable number of walls with labyrinth door, et cetera, to maintain the doses outside, you know, as low as reasonably possible.

MEMBER RYAN: And one other thing you might want to think about as you're going through, and point out, maybe where this is the case.

And I appreciate the compartmentalization comment you just made, but there's an upside which is, you know, isolation, but there's a downside which is if there ever is a situation where one of these compartments does become contaminated or significantly contaminated it would create a real clean up headache.

They can be a real decontamination challenge to do it in confined spaces and so forth. So, maybe you can talk about the balancing act between those two issues of decontamination and ease of contamination with removing this stuff versus the advantage of worker exposure management with fixed compartmentalized facilities.

And I have seen extremes in both directions, so maybe you could help us understand how you made that -- you know, those judgments.

MR. PEREZ: Yes. And I think I'll get to

that when we talk about building --

MEMBER RYAN: No problem. I just wanted to point that in your head until we get a --

MR. PEREZ: And I took a note, so we won't forget.

MEMBER RYAN: Okay.

MR. PEREZ: And then as we all know, there is industry experience with environmental contamination, tritium. That has also been looked at, and we have design features that I will mention later.

MEMBER RYAN: Okay.

MR. PEREZ: That provide, you know, protection against environmental contamination.

MEMBER RYAN: So you have really specifically focused on the tritium task force --

MR. PEREZ: Yes.

MEMBER RYAN: -- and all that -- all the resulting NEI documents?

MR. PEREZ: Yes.

MEMBER RYAN: And so forth on that. Okay. Thank you.

MR. PEREZ: Yes. We participated with NEI on those task force meetings.

MEMBER RYAN: Great.

MR. PEREZ: And again, I would also like

to mention ALARA is applied in the time process. We have an ALARA design guide. Our system engineers work closely with my group. We do ALARA reviews.

We provide training to the system engineers and component engineers, the folks that really own the systems, and we're bringing the awareness level continuously through the design process, and we follow whatever is happening in industry to continue to learn from those experiences.

Let me make sure I don't lose track of -- Within the final safety analysis report, when we performed the shielding evaluations basically, the occupational exposure calculations, we came up with two source terms.

The first source term is for normal operations. It's what you expect to see during a normal plan operating cycle. A second source term was generated for accident conditions.

And those, of course, are used in the cases of a postaccident access, and those are NUREG 0737 accessibility mission doses.

The source terms will determine both contained and airborne sources. Contained sources are simply the liquid inside a pipe, which provides basically a gamma, you know, a gamma shine from that pipe.

And the airborne could be from leakage of such systems that would flash and create an airborne contamination which could, of course, can also be something like tritium could also become easily airborne, okay, if not properly controlled.

For normal operations, we use the guidance in the 10 review plan, Chapter 12. As I mentioned, from there you can derive these two types contained in the airborne.

The radiation sources are calculated and dose assessments are performed to ensure that our occupational doses are ALARA, and also to ensure that the shine from the facility, okay, is maintained ALARA. In other words, just the exposure -- air exposure from the facility.

MEMBER RYAN: You mean at the boundary?

MR. PEREZ: At the boundary, exactly.

MEMBER RYAN: Yes. Okay.

MR. PEREZ: For the EPA -- right.

MEMBER RYAN: Okay.

MR. PEREZ: We start with a failed fuel fraction of 0.25 percent. That's within the SRP, except that for the iodine and the noble gases, we brought them up to match the technical specifications limiting condition of operation, which is one microcurie per gram iodine 131, dose equivalent and 210 for xenon 133.

CHAIRMAN POWERS: Maybe you can explain to me. I've never understood how you take iodine 131 dose equivalent.

MR. PEREZ: You basically take into account the dose conversion factors and you look at the concentrations for all of the iodine constituents that when summed up, taking into account the dose conversion factors, give you the equivalent iodine 131 concentration.

MEMBER RYAN: That's kind of like the medical guys in the old days would express milligrams of radium equivalent when they used the cesium source for radiation therapy.

CHAIRMAN POWERS: I didn't understand how they did that, either.

MEMBER RYAN: Well, that's because they only know how to measure radium, so they had to make everything radium equivalent. So, it's -- I mean, we're probably sophisticated enough we could handle all of the iodine isotopes by themselves, but we don't.

MR. PEREZ: It makes it easier, I think,

in a technical specification to have one number.

MEMBER RYAN: It makes it simpler to calculate.

MR. PEREZ: Right.

MEMBER RYAN: But it makes it harder to understand.

MR. PEREZ: Yes.

MEMBER RYAN: Particularly in the days of, you know, computers that can do everything.

MR. PEREZ: Yes. And by using this failed fuel fraction, which equates to about 160 failed fuel rods for the EPR, we also verified the fuel performance, for the AREVA PWR fuel, is well-founded by this assumption.

MEMBER RYAN: What is that performance level? What is the failed fuel fraction you expect?

MR. PEREZ: Expected for this plant, I am going by historical.

MEMBER RYAN: Yes.

MR. PEREZ: Okay. Which is less than a hundred. The number right now I can't remember, but it's --

MEMBER RYAN: Roughly two-thirds of the 160?

MR. PEREZ: Two-thirds, exactly.

MEMBER RYAN: So -- yes.

MR. PEREZ: And what I used was not only AREVA data, but the info tracking that is done.

MEMBER RYAN: When you look at failed fuel, you're looking at a leaching source term?

MR. PEREZ: Okay. That's also an interesting subject, okay, because different -- different folks can interpret how you do it different. I would like to share with you what we did.

We take the core inventory, okay, all of the core inventory, multiply that times this failed fuel fraction of .025 percent, and then apply the radionuclide-specific appearance rates, okay, from the fuel. And that is how we come up with the source term introduced into the reactor coolant.

MEMBER RYAN: What's the appearance rate exactly?

MR. PEREZ: Okay. The appearance rate comes from NUREG 0017, which is the old GALE Code that gives you the number of microcuries of activity appearing per second in the radionuclide groups, the halogens, alyloids, et cetera.

MEMBER RYAN: Okay. So you're taking it from a NUREG forward. Have you looked back to see what the basis of the NUREG might have been versus today's practice?

MR. PEREZ: It is almost archeology. Okay.

MEMBER RYAN: Well-said.

MR. PEREZ: You go back -- and seriously, going through those, I have tried to do that. I have used services to try to come back and what I can only arrive at is if you recall NUREG 0017 was based on real operating plant data in the late -- maybe the late or mid-Seventies.

There was worse fuel performance during those days.

MEMBER RYAN: Sure.

MR. PEREZ: The library within that GALE reflects bad performance. The GALE 0017, Revision Zero -- not the latest revision, but Revision Zero, stated that that source term corresponds to about 0.125 percent failed fuel.

So I wondered if it was backed out, if somehow numerically these were backed out. That's all I can -- that's all I can --

MEMBER RYAN: But you really can't reconstruct it forensically by searching the documents.

MR. PEREZ: No.

MEMBER RYAN: Is the end of the story. You can go with what the statement basis is but there's no confirmation.

MR. PEREZ: That within the available documents that I have, that's correct.

It's interesting that in that NUREG 0017, the conclusion was that this failed fuel was 0.125, the SRP has 0.25. They multiplied it by two. It was multiplied by two? I don't know.

MEMBER RYAN: Sure.

MR. PEREZ: Okay. But all I can say is, based on industry performance, based on AREVA performance, this assumption, for the purposes of shielding is a bounding assumption today.

MEMBER RYAN: Yes. I think that's the take-away messages that your .25 fraction by any reckoning seems to be, you know, to have a margin from what --

MR. PEREZ: Yes.

MEMBER RYAN: -- from what is likely to occur with current fuel and the current circumstances, that's your conclusion and it seems reasonable. It would be nice to have more forensic study or archeology, as you say, to sort it out.

But if I recall, Dr. Powers, there's a

couple of research projects underway on the GALE Code and updating the information contained in it, as I recall. We looked at it on the research report.

CHAIRMAN POWERS: Well, I'll have to look at that. What you're telling me is that we do not have -- I think what you're telling me is that we do not have an archival publication that says that we took these clad fuel with a split in it and let it leech for under X conditions for Y periods of time.

MEMBER RYAN: Not one that's readily accessible to put your hands on.

CHAIRMAN POWERS: Yes. Archival publication.

MEMBER RYAN: Right.

CHAIRMAN POWERS: That's remarkable, isn't it?

MEMBER RYAN: Yes, sir. And we had -- I mean, it's --

CHAIRMAN POWERS: I mean, it's not their fault.

MEMBER RYAN: No, no, but -- but again, I mean, I think from the standpoint of the discussion at hand that the fact is that current fuel data shows that the assumption of .25 percent is not testing any value --

CHAIRMAN POWERS: Yes, I mean they're --

MEMBER RYAN: -- that's fine.

CHAIRMAN POWERS: They're doing --

MEMBER RYAN: But I think it goes on the list of things that we ought to explore a little bit further and, you know, what -- are they addressing this in the work that's underway on the GALE Code.

CHAIRMAN POWERS: Well, I think the -- I mean, I think that's an excellent point. I think we also have to look -- we have to presume there to be a substantial uncertainty on all kinds of numbers we're getting out of this.

MEMBER RYAN: Well, the other point about the GALE Code is it's written on Fortran 4 and there are very few comments in the code listing. So, you know, that is a forensic study right there.

CHAIRMAN POWERS: Must we start over on this area? Okay. Please continue.

MEMBER RYAN: Anyway, that's -- but again, I applaud the approach you're taking, is you really are trying to make sure that you have some sense that you have a margin from what is likely, and that's more recent fuel performance, and at least in my opinion is a better measure of where we'd be today. So, thank you.

MR. PEREZ: Thank you. Together with these failed fuel source term which again, are fission product, you also have activation corrosion products, and you also, of course, have your nitrogen 16, which is a very strong gamma source in the recirculating RCS.

So, all these constitute the reactor coolant system shielding source term inside the reactor building.

MEMBER RYAN: How did you look at fission product and activation product over time? Did you look at fuel burn-up and, you know, fuel reloads and all those kinds of things over time?

MR. PEREZ: What we did for the fission product, we looked at a range of enrichments. I believe it was like -- I think it was from two to five weight percent, and then we took burn-up steps from five all the way up to 62 in steps of maybe ten gigawatts with image of time.

That provided us a, if you would, a sources of core inventory from which we could take bounding values of radionuclides independent of where they are.

So, in other words, is a composite source done that will bound any operating strategy.

Corrosion products come from ANSI 18.1. And again, there is better -- I think the industry has better performance now from when 18.1 was last revised which I think was 1999, I think -- I can't remember.

But that's where the corrosion activation product concentrations came from.

MEMBER RYAN: Okay. Thank you.

MR. PEREZ: For normal operations inside the atmosphere of the reactor building we also look at argon 41, basically neutron activation of argon in the air. Tritium is also addressed in the reactor cooling system.

The secondary coolant source term is derived from the transfer of reactive coolant inventory to the secondary side through assumed steam generator defect, and the leakage rate is the tech spec maximum.

So when you transfer radioactivity from the reactor coolant to the secondary side, the radioactive effluents are processed by the gaseous and liquid waste processing systems.

So, as you can see, we have radioactivity traversing through the plant out to where they are ultimately processed.

CHAIRMAN POWERS: And you know the

performance of these systems how?

MR. PEREZ: We know the performance of these systems based on, for example, ion change resins, there's lots of information from EPRI on the different media for, you know, resin media targeting to remove cesium or iodine or other nuclides.

CHAIRMAN POWERS: I would be surprised if the ion exchange resins have a permanent equivalent to the database.

MR. PEREZ: I agree. They are not permanent. Improvements in chemistry are continuous. The data that you're presented is a snapshot in time, basically 2005 is when we submitted. Correct?

MS. PEDERSON: Correct.

MR. PEREZ: So these analyses are from 2005. The gaseous waste products --

MR. PEREZ: Thank you. 2007. Thank you.

The gaseous waste processing system, we calculated that delay times for nobel gases, krypton and xenon, those are calculated from the mass and the characteristics of the system.

Other data -- and again, we were

talking a little bit about Chapter 11, the data on how the liquid waste components work, come from field data. What went in, what came out, basically gives us a DF, and that's how we modeled in Chapter 11.

MEMBER RYAN: And I guess it will really come down to the comfort level you have with whatever the configuration is for the system and the design versus the system that you have real data for. How do you feel about that alignment?

MR. PEREZ: Okay. We talked to vendors, and we talked to vendors and said, this is the source term that we plan to send to your -- to your media. What can we throw at you, what do you have?

And again, no one said we can't do it. They said, oh, you need to put this media and then this media and then the other media. So, for the .25 percent failed fuel, they saw radionuclides that don't appear in industry by far.

MEMBER RYAN: Yes.

MR. PEREZ: And these gentlemen were from, you know, supporting operating plants, not design certification applications, so they were asking where these numbers come from.

You explain to them the failed fuel fraction and then they say, okay, we can go through -- we know when plants have failed fuel. We know where these systems operated. We can look at that performance.

And I'm confident, again, in my Chapter 11 that the values used for clean-up rates or DS can be supported today, okay, with specific media. And there are vendors out there right now publicly providing, you know, this -- you know, this targeted media for whatever planned effluent, you know, requirements.

So, we have benchmark with industry.

MEMBER RYAN: Here and in Europe or --

MR. PEREZ: No, here.

MEMBER RYAN: Just here?

MR. PEREZ: Yes, here. And with Europe, but what I just spoke about, about myself speaking with a vendor, was a US vendor.

MEMBER RYAN: Okay.

CHAIRMAN POWERS: There is, I mean, as far as ion exchange resins and things like that, the United States is still pretty limited in that field.

MEMBER RYAN: Yes.

CHAIRMAN POWERS: The biggest problem that they're really going to have on those materials are counterfeit supplies, mechanical material.

MEMBER RYAN: Yes.

CHAIRMAN POWERS: It's going to be a big -- I mean, I just anticipate that it will be a headache because the -- I mean, whereas we have the technology, we don't actually make them here in this country anymore, and so getting counterfeit will be a headache.

MEMBER RYAN: But that falls under QA or something like that. Please go ahead.

MR. PEREZ: The spent fuel pool is also a radiation source due to an assumed fuel defect that we assign to the fuel in the pool. The corrosion activation products and, of course, the tritium that is transferred through the fuel transfer canal.

The radiation doses are calculated and dose assessments, are performed to ensure occupational doses remain ALARA and the sources are also used in the design of ventilation systems

-- and let me go to the next line.

And for accident conditions, the radiation sources from the radiological bounding accident, the initial -- the initial concentration -- I should say the initial concentration, and the reactor coolant system for the loss of cooling accident is from the .25 percent failed fuel.

That is the initial. And then, of course, on top of value, add the accident. So, you will see Chapter 15 and Chapter 12 referencing back and forth to Chapter 11 where the sources are derived.

I should mention here that the accident source term is based on the alternative source term, designed by such coolant accident, and that source term is used for the postaccident shielding evaluation.

That means the recirculating fluid from the ECCS, emergency core cooling system contains a source term -- a source term driven from the design basis loss of cooling accident that uses the alternative source.

And that is done to evaluate, you know, mission doses and again maintain both ALARA and meet, of course, the accepted rates.

MEMBER RYAN: What do you assume about the core in that case?

MR. PEREZ: There's a nonmechanistic core overheat.

MEMBER RYAN: Yes, but I mean in terms of the inventories, you've assumed a worst-case inventory?

MR. PEREZ: Is the same -- it comes from that same bounding that I described before, that core inventories, I'd say more.

MEMBER RYAN: Okay.

MR. PEREZ: So we are trying to -- we've tried to really have one core inventory, and from there derive different source terms depending on the type of evaluation you're doing.

MEMBER RYAN: Got you. Thank you.

MR. PEREZ: Again, as I mentioned, in this case for accident conditions, that source term applies to ensure mission doses are ALARA. We looked at the contained sources in the ECCS, just gamma shine through a shield, and also the airborne source from the new safety features. ECCS gas leakage is assumed in the analysis.

CHAIRMAN POWERS: You used the NUREG 1.183 source term?

MR. PEREZ: Yes.

CHAIRMAN POWERS: That's derived at NUREG 14.56.

MR. PEREZ: Yes. 65, yes.

CHAIRMAN POWERS: And that document specifically excludes fuel with burn-up's greater than about 40 gigawatt days per ton --somewhere around there is -- I mean, you're going to keep your fuel limited to less than 40 gigawatt days per ton?

MR. PEREZ: No. We -- a 40 gigawatt days per metric ton represents for us a core average burn-up. Okay. We can take an assembly out to, for example, 62 gigawatts per metric ton. Not every assembly is a 62. Okay. Not every assembly.

If we do a fuel-hanging accident we will take a 62 gigawatts per metric ton burn-up assembly. If we do a shielding evaluation when the assembly goes through the transfer tube, we will use that highest indicator that we can.

CHAIRMAN POWERS: So, do you have to recalculate -- I mean, what I'm trying to understand is, suppose you're doing a fuel-handling accident with a 58 gigawatt per day ton assembly, what do you use for the source term.

MR. PEREZ: It comes from the parametric study that we performed. We take the core inventory for that burn-up step. Because it's a fuel-handling accident we have to apply a peaking factor. We apply a peaking factor --

CHAIRMAN POWERS: Okay. So you use a depletion code of some sort?

MR. PEREZ: Yes. Yes. Everything starts off with -- you know, with origin as the base of the depletion.

CHAIRMAN POWERS: That's all I was fishing for.

MR. PEREZ: Okay. In 12.3 we actually performed the occupational doses and the off-site external doses to basically demonstrate that we have adequately addressed ALARA with our specific design features.

And here, what we've looked at is the -- for example, the physical plant layout. I mentioned compartmentalization creates permanent shields.

And, Dr. Ryan, you mentioned you wanted me to come back and talk a little bit. I did not mean to say that we were creating anything like confined areas. These compartments are actually quite large.

And what they are actually doing is, in fact, if they are contaminated for some reason, there are design features to capture berms, for example, to capture any possible leakage and put it into the plant vent and drain system.

These areas where the activity is more concentrated will have these ventilation cells. So, you're segregating that area, and if you have to go clean it up, which you will, in case it's contaminated, you can prevent spreading further the contamination because it's contained within this box, and it's not a confined space. Actually, it's a room. So, it's a large room.

MEMBER RYAN: I imagine they range in size. Can you give me some idea of the floor space range for -- of these compartments? Are they ten-by-ten up to a thousand-by-a thousand or are they --

MR. PEREZ: Well, what I can do, within the FSAR -- and again, I couldn't put these pictures on the screen because they're SUNSI information.

MEMBER RYAN: I have all of the

Chapter 12 figures here. If you could point to one that's in 12, that would be great.

MR. PEREZ: For example, 12.336.

MEMBER RYAN: 336.

CHAIRMAN POWERS: If this information is proprietary, you don't have to -- I mean, the problem is we have to make it publicly-available if it's asked for.

MEMBER RYAN: Okay.

CHAIRMAN POWERS: Unless we go to closed session.

MEMBER RYAN: Yes. Maybe we can decide on doing that later if we need to.

CHAIRMAN POWERS: Okay.

MEMBER RYAN: But if you could just give me the range of the floor spaces, that's a good start.

MR. PEREZ: Just looking at this drawing, it could be like 20-by-20 feet.

MEMBER RYAN: That would be a smaller space?

MR. PEREZ: Here, from just my having it upside-down --

MEMBER RYAN: That's fine. I appreciate that.

MR. PEREZ: I would say it's a small space.

MEMBER RYAN: A smaller one and then they go up to what size? A hundred-by-a hundred?

MR. PEREZ: I would say yes.

MEMBER RYAN: Okay.

MR. PEREZ: The ECCS rooms are quite large.

MEMBER RYAN: Okay.

MR. PEREZ: Yes. So you're not -- you know, you're not at all in a confined, difficult-to-get-at, you know, area.

MEMBER RYAN: Well, and the difficulty, again, I point out is probably more important to my question.

MR. PEREZ: Yes.

MEMBER RYAN: I've seen process facilities where things are intentionally compartmentalized and very hard to reenter once they're sealed up. And that's not what we're talking here?

MR. PEREZ: That is not what we are talking about. In fact, it is the opposite. Those components that we know need serving --

MEMBER RYAN: Yes.

MR. PEREZ: -- are placed in locations that's readily accessible. Okay. So, for example, I have a bullet here, and I may be jumping ahead. I have a bullet here that within the design we've eliminated extensively the need for temporary scaffolding by providing permanent platforms, you know, for people to go up and work at, and not receive dose as they are erecting a platform.

Or, we have provided components that need servicing closer to the floor. Okay.

MEMBER RYAN: Yes, that's good stuff.

MR. PEREZ: So, with some -- there was -- maintenance was considered in the design of the plant. Outage activities was considered.

And again, in the FSAR, these drawings show, for example, lower-dose areas are always provided before a high-dose area, so you gradually approach a high-dose area.

MEMBER RYAN: So the staging and preparation can go on and that sort of stuff then becomes a low-dose activity --

MR. PEREZ: Yes.

MEMBER RYAN: -- as opposed to a high dose.

MR. PEREZ: Exactly. So you can do a prejob brief right outside, if you would, where the job is being done.

MEMBER RYAN: Got you.

MR. PEREZ: If you need to come out to rest you are right outside. You don't have to walk out, you know, to the checkpoint.

MEMBER RYAN: Yes. That would be helpful to kind of calibrate, too, just so we all have the same numerical understanding. What's a low-dose area versus an intermediate versus a high? Do you have a cut there?

MR. PEREZ: I have a slide earlier and I'll present it.

MEMBER RYAN: Okay. Fine. Okay. Terrific.

MR. PEREZ: The reactor building is unique. You have a two-compartment reactor building where the reactor vessel, the steam generators are pressurized. Where the N-16 is, where you have the greatest concentration in the reactor cooling system is inside the innermost compartment, the equipment compartment with it's own ventilation system.

And outside of that you have

accessible areas for personnel to go in at power and maintain doses ALARA dose -- you know, F-1 dose -- from F-1 and from shine.

Just looking at the amount of concrete that's there will give you an appreciation for the dose reduction. So you can go in at power with relatively low doses, 10 millirem or less, okay, anywhere in this accessible area.

And this is, again, with the assumption you have this failed fuel fraction. So, in reality it will be even less of a dose rate.

MEMBER RYAN: What -- in that area I mean, I guess you're going to make assumptions on -- to get here, but what's the internal exposure versus external. Is most of it external?

MR. PEREZ: Yes. Yes. It's all BS. I mentioned the fact that we have four safety ejection trains, ECCS trains, and that, having more equipment you can say, well, that creates potentially more dose because you have to do more maintenance.

But now you can be selective when you do that maintenance, because you can take one system completely off-line, inoperable. You have three others and you can still meet single failure.

You could now have the luxury of time to allow preparation of the area to get the doses to where you want to get them without having to rush because you have an eight-hour LCO at 12 hour, you know, LCO.

I mentioned about the radiation zones are graduated. You will see -- you will see that in a second. And again, the goal is to maintain general areas, general access area throughout the plant to be less than or equal to two and a half millirem per hour.

So, these are areas where you have always more staircases, elevators. So, we want to keep those areas where we call green area, two and a half millirem or less.

So, only when you get that staging area, getting close to doing the work would you receive, you know, and exposure that requires monitoring.

Here is a picture that has been -- has been basically sanitized for public viewing, but I would like to show you a couple of important points.

This also, Dr. Ryan, gives you an idea of the compartmentalization, all these little boxes with concrete and rebar surrounding it provides shielding.

The spent fuel pool is the -- yes, thank you. Something interesting here, for PWR, here is a spent fuel pool. Notice that it is -- that it's not at the lowest elevation in contact with potentially the ground.

It's totally surrounded by inspection, inspectable areas. It has leak detection. If you were to have a leak in this pool right there, it cannot directly go into the ground. It has leak detection and, again, it is totally surrounded by areas that you can inspect.

I'm going to try to walk through. If you look at the staircases out here, this is the area where you have access all the way up here at power. Let's see what else I can point at.

Here again, the massive shielding provide external doses out here that are less than one millirem.

MEMBER RYAN: All the light blue is water of some kind. Would you point out what those other sources are?

MR. PEREZ: Yes. In, for example, I mentioned the spent fuel pool.

MEMBER RYAN: Oh, yes.

MR. PEREZ: Sometimes when you do these cuts I get a little --

MEMBER RYAN: Yes.

MR. PEREZ: Okay. You have -- and someone can maybe help me. This is called the in -- in containment. Thank you. In containment refueling water storage tank, IRWST.

Is this the emergency -- excuse me. Emergency feedwater. Okay. Thank you. And again, within the spent fuel pool you have new fuel in, you know, two pools.

Here in this cut I believe the same as this, except you're looking at it from a different cut.

The sources radiation, again, in the water, the spent fuel pool, the refueling, you know, cavity area, you have here, of course, the reactor vessel with the recirculating coolant.

This is what I meant by radiation zones and graduated zones. Our initial -- our initial idea was to make this color-coded, and now that's created more challenges than I ever thought, but the idea was, you go from green on the right, and that green is two and a half millirem rads, and that's what I meant by the staircases here or elevators or areas of general access will be two and a half millirem or less.

You gradually go to a yellow area which is ten millirem or less. I'm sorry, 25 millirem or less. Magenta, 100 millirem or less. Finally, the red area is greater than a hundred.

These are based on, again, a quarter percent failed fuel fraction, --

MEMBER RYAN: Millirem per hour.

MR. PEREZ: Millirem per hour, yes. Millirem per hour.

Within the FSAR there's a color table that gives you the breakdown of the -- of the ranges. And I just took one snapshot from, you know, from the FSAR figures.

I mentioned the reactor building contains the containment building itself and the shield building with this annulus space which again is a -- is a considerable amount of concrete that keeps the external doses very, very low.

It has those two compartments segregating the potential radioactive sources from the area that's accessible with separate ventilation system, and other design features that help reduce occupational doses.

The EPR has permanent cavity seal for refueling activities, so the dose required to install a, you know, a bladder that you have to fill up with air to seal that cavity doesn't exist. It's permanently installed. It's there, so now that dose contribution is gone.

The safeguard buildings, there are four separate safeguard buildings, housing independent divisions of emergency core coolant. Each building is divided into radiological control area and an uncontrolled area.

System containing radiation sources are placed closest to the reactor building, and at the lower two floors. And now we get benefit of being below grade, getting shielding from the ground.

And keeping the systems close to the reactor building, reducing -- reduces the potential spread of contaminations throughout other systems.

MEMBER RYAN: Just back here in slide nine on the drawing, if I may, it looks like there's two little spots, one on each side. Is that the ground level where that little --

MR. PEREZ: Yes.

MEMBER RYAN: Okay.

MR. PEREZ: Yes. Thank you for pointing that out. Yes.

MEMBER RYAN: One point to catch as we go along, is it the 1406 requirements, NEI 08.08 and all that, could you begin to pick up on that aspect, because I'm sure we're getting into areas that address that as well.

MR. PEREZ: Yes. We will be getting to those areas.

MEMBER RYAN: And I'm getting out of order. Just stick with your order. Go ahead.

MR. PEREZ: I'll stick with my order.

MEMBER RYAN: Okay.

MR. PEREZ: Let me -- thank you.

MEMBER RYAN: Sure.

MR. PEREZ: Within the safeguard buildings you have two different ventilation cells, one for the lower levels, one for the upper levels. The two atmospheres don't mix.

So, if there is some leakage due to ECCS leakage, you're not going to have an airborne contaminant in the upper elevations. So we are segregating again, putting everything as low as possible.

There's only one exception, and that's the gaseous waste processing system that those delay beds are at a higher, you know, elevation. But everything else is low and close to the reactor building.

You know, the fuel building, again, is divided into cells with ventilation and isolation. Location of the fuel pool, as I mentioned, from that drawing, eliminates the possibility of having a direct environmental contamination as mentioned in 10 CFR 20.1406 and, as we have seen from operating experience where you have a leaking spent fuel pool going into that -- going into the environment.

So, here's a design feature keeping the pool away from a direct pathway. Okay. So, that's one 10 CFR 20.06 feature. I'll mention -- I'll mention more as I go forward.

In the auxiliary building, again, there are three ventilation cells. You're trying to, again, minimize -- which is again 10 CFR 20.1406, the potential spread of contamination in the facility.

Radioactive waste processing where you have a good potential for airborne and shine -- and spills, you have -- I'm glad you mentioned spills, because you have berms to catch whatever could spill.

You have strategically-located tanks staying to one area so you limit, okay, the spread of the spill, collect it into the vents and drain system. It's an automated process where the operator is in a shielded control room handling remotely cranes and devices to manipulate waste.

MEMBER RYAN: Is the design such that all the waste processing equipment is part of the design as opposed to services procured from vendors or --

MR. PEREZ: Right now the only -- the rates of demineralizer described in the design certification package, that is a vendor-supplied device. And as we mentioned earlier, technology changes, so you want -- you know, you want to continue to support -- you know, to have whatever is the latest.

The design does include a minimum DF per nuclide group that has to be met.

MEMBER RYAN: Yes. Now, that's for an installed unit. I'm talking about being, you know, outage services, and people that come in and do clean-up of the systems and then take away or exchange resins and all of that.

MR. PEREZ: No, within design certification that's not -- that's not within design certifications.

MEMBER RYAN: That's going to be a separate activity or --

MR. PEREZ: Yes.

MEMBER RYAN: I guess you're envisioning that all the waste processing to produce a final waste product is built into the plant?

MR. PEREZ: Correct. For design certification, everything that's described there is a complete system. Yes.

CHAIRMAN POWERS: Have you done anything on surfaces for -- to facilitate clean-up of spills and things like that?

MR. PEREZ: Yes. I am not a coatings person, okay, but work has been done to specify a coating to seal the concrete and to facilitate a clean-up of those areas. So I know --

CHAIRMAN POWERS: Epoxy, polyurethane? What is it?

MR. PEREZ: I could not answer that question. And I think Chapter -- or 11.

MEMBER RYAN: Maybe we could just have that as a take-away question --

CHAIRMAN POWERS: Yes. Just assign that to Sandra till our --

MS. SLOAN: There we go. But this is about coatings?

MR. PEREZ: Yes. Let me just -- if you want to shoot an email --

CHAIRMAN POWERS: Or just tell her 12 pages will be plenty adequate with detailed drawings and design specifications.

MR. PEREZ: And she will have to apply it.

MEMBER RYAN: Yes.

CHAIRMAN POWERS: That would be good. A video of her actually applying the coating material would facilitate my understanding of the issue so much more clearly. It would be really beneficial.

MR. PEREZ: Okay. We'll try to get an answer to that.

CHAIRMAN POWERS: I would appreciate it.

MR. PEREZ: And again, as I mentioned, the waste processing in the waste building, if you look at that waste building, there's a lot of segregation keeping, you know, the waste away from the environment and the waste away -- and the exposure from the waste processing away from the operator.

MEMBER RYAN: Of course, the $64,000 question is when you generate waste packages, what are you going to do with them. Yes?

MR. PEREZ: Yes.

MEMBER RYAN: Yes.

MR. PEREZ: The answer is yes, that is a $64,000 question, yes.

CHAIRMAN POWERS: I thought it was $640,000, at least.

MR. PEREZ: What we have done is, again, working -- this is within the design certification, working with our COL Applicant, those have been quite a few RAI's. It depends what state you're dealing with.

MEMBER RYAN: Yes.

MR. PEREZ: We have storage capacity

of at least five years. Okay? It will be for Type B and C waste. But, you're correct.

MEMBER RYAN: So, I mean, at some point, I guess I want to guess you're going to at least considering a waste building on some source term for shine and all that on it's own, or no?

MR. PEREZ: Design certification, no.

MEMBER RYAN: No, but for COL, yes.

MR. PEREZ: For COL, depending on site, yes.

MEMBER RYAN: I got you.

MR. PEREZ: Yes.

MEMBER RYAN: Okay. Thank you. And that makes sense that it would be a COL issue.

MS. SLOAN: I'll let you know.

MR. PEREZ: Thank you.

MEMBER RYAN: Just a quick question, if I may, back to the design page on nine. And it's really, you know, it's not so much design certification as maybe design certification and COL.

Where the groundwater exists at a given site is a big impact on where -- what you're going to do down at the base.

MR. PEREZ: Yes.

MEMBER RYAN: Do you have some options here you can include in the design SER or how does that work?

MR. PEREZ: Okay. Well, we -- how it works is as follows: In Chapter 2, and I want to say it's 2.14, I can't remember right now, where you have to look at breaking the liquid tank that has the highest concentration radioactivity, and look at the environmental effects of, and this deals with Regulatory Guide 1.143 for classifying radioactive equipment.

We took a -- one of the COL sites, as soon we broke -- I think we broke more than one tank. I remember that because the volume was considerable. And then you look at whatever KV values, the migration of nuclides through the environment, we use conservative values, and we showed that the concentrations met 10 CFR 20, Appendix bravo, table 2, concentrations.

It becomes a COL item for them to then perform their own calculation for the site-specific analysis.

MEMBER RYAN: And 10 CFR 20, Appendix B may not be the numbers that they have to meet based on the state.

MR. PEREZ: Yes. It depends on the state, yes. And again, the -- we did the analysis as a demonstrative analysis.

MEMBER RYAN: Yes, I understand.

MR. PEREZ: For the COL applicant then to perform the site-specific -- the site-specific case.

MEMBER RYAN: Okay. Thanks.

MR. PEREZ: All right. When we started the design certification, FSAR preparation, 10 CFR 20.14.06 in the industry was quite popular, so we quickly started paying attention and we looked at our design features and we looked at enhancing the design features.

I keep talking about compartmentalization. That helps us minimize the spread of contamination, and potentially contaminated systems are isolated from clean systems by two or more isolating features.

The staff asked us through RAI's very direct questions to demonstrate the barriers between systems. And in some cases we did a design change to add a barrier. In other cases we found no, we had sufficient barriers, keeping a potentially radioactive system from a clean system getting cross-contaminated.

We have component leak detection wherever there is a direct path to the environment. So, within like the concept of a pipe within a pipe with leak detection, a double wall, some at the lowest elevation. Okay. Again, with leak detection so you can start, you know, corrective actions if you start noticing a leakage.

MEMBER RYAN: The leak detection technology which, again, I'll plead ignorance on being anywhere knowledgeable enough about to ask a good question but, you know, when you think about a plant operating for 20 years, 40 years, 60 years, pick a number, you've got a detection technology inside, you know, between two pipes.

How reliable is that going to be over the long haul and are there provisions to maintain those kind of things or what happens if they fail, what sort of in-place testing are you going to think about?

It's a great idea, but I'm just wondering where's the reliability factor for that?

MR. PEREZ: You know, you're correct. I mean, I'll tell you right now, on the floor of my home by the water heater I have a leak detector, and then every now and then it goes off for no reason. Time to replace it.

MEMBER RYAN: Right.

MR. PEREZ: These locations will also provide accessibility, okay, so they can be -- it can be replaced and maintained.

MEMBER RYAN: So, service and replacement is something you're planning into the engineering for these --

MR. PEREZ: Absolutely.

MEMBER RYAN: Okay.

MR. PEREZ: Because it is a design feature that has to be maintained.

MEMBER RYAN: Right. Okay. Good.

MR. PEREZ: I mentioned about the spent fuel pool away from external walls and floors.

Then I want to talk about -- I want to talk about one -- one effort that we did that I would like to show with you because I think it demonstrated a proactive approach to 10 CFR 20.1406.

Together with the -- our licensing folks, we looked at looking at industry experience, how did the tritium get out from the plant. Do we have a potential for doing the same?

One area, loss of outside power, using atmospheric dump valves or if you lift a steam generator safe -- you know, safety relief valve, the condensation from the steam leaving will be -- some day, will contain tritium.

And the question is where does it -- how do you collect that condensation. For the EPR we actually have silencers on the safety relief drain exhaust. The silencer actually produces more condensation.

So, we noticed, hum, the way this is running right now, it looks like it is going into the parking lot. Not a good idea.

We did a design change. We worked with the system engineer. Now we're routing it into rad waste.

We had to re-look at sizing tanks. We had to re-look at, you know, how much flow.

MEMBER RYAN: How's the rad waste system going to stop tritium?

MR. PEREZ: No, but it's controlling where you release it from.

MEMBER RYAN: So you basically accumulate it in the rad waste system?

MR. PEREZ: And release it through your permitted release point, meeting 10 CFR 20, Appendix B, meeting state NPDES permits, because right -- I mean, as long as you are permitted to release that waste you're okay.

MEMBER RYAN: Yes. And, you know, an expected and controlled and plan release --

MR. PEREZ: Yes.

MEMBER RYAN: -- is what's required, not a whoops, what's this over here on the step.

MR. PEREZ: Exactly. Exactly.

MEMBER RYAN: But in that system you designed, I guess to accumulate a large enough volume that you haven't stressed any other system.

MR. PEREZ: Not only accumulate it, but go ahead and try -- it provides a collection tray --

MEMBER RYAN: Yes.

MR. PEREZ: -- transferring it to the rad waste system.

MEMBER RYAN: Okay. And the Rad waste system is sized so that that's a new source for a rad waste system.

MR. PEREZ: Exactly. And everything had to be looked at --

MEMBER RYAN: Okay.

MR. PEREZ: You know, from that plant, and in that we had system engineers involved from day one.

MEMBER RYAN: Okay.

MR. PEREZ: So, that's an example. We are, to date, continue to look, okay, as we move forward, detail design. We continue to look. And, you know, we will monitor what's happening in the industry, look at our design and address any changes that we may need.

MEMBER RYAN: Yes, sir. Thank you.

MR. PEREZ: With new radiation protection, we thought, with all this -- well, let me go to the next slide because I think I have a picture. Let me continue and then I'll talk about another -- another lesson learned here.

Once we have the source terms, we know where they are located, we have the geometries. Now we can do dose calculations. We can do dose rate calculations. We have sources and geometry.

It's a big plant. Fortunately a lot of pipes work the same, so you can have a normagram for an RCS liquid source term for eight-inch pipes, 12, 15, whatever, and you get dose rates from them.

You have four ECCS trains. Well, we looked at dose rates in one area and just repeated, okay, those dose rates for the other areas. Some unique cases, volume control tank, you have a vapor space on the top and liquid on the bottom.

You have six walls, six surfaces, four walls, a ceiling and a floor. You do the dose rate all around and you look at other contributing sources to that other area.

Another unique location is the fuel transfer to -- because during the refueling outage you have spent fuel pool traversing the annulus where people could be working.

So, we have a discussion with the staff where a reviewer asked us about how do you control access, and well, we control access to the entire annulus space.

Well, that may not be the best thing in an outage because you may be paralyzing other activity. So, in answering that RAI we came up with a design change, okay, to put, you know, an access barrier localized so, you know, you are controlling doses, but you are not limiting outage activities.

So, again, that was a learning experience. I will mention that every now and then when we learn these items, such as the MSRQ release and the condensation, I have calls with my colleagues in Europe, because our job is to make sure these plans, wherever they may be, okay, are -- you know, are maintaining doses, ALARA, minimizing the potential for spread of contamination.

So, we have kind of a cross-pollination, and we learn from them. They learn from us. So, that's another example and I'd like to show you -- okay. Let me finish with this.

Then from these dose calculations we come up with the radiation zones that you saw that were color-coded, and from those zones you can then calculate exposure to personnel performing activities.

The next slide shows the area that I just mentioned. The blue is water. And here is where the fuel assembly will traverse. The annulus space, you see a very substantially thick labyrinth, another labyrinth here, and before there was no gate and I had to limit -- I had to limit the picture just to this because there were no gates allowing access here.

So now, now we do a design change, add a gate. So, that's an example of a unique case where more detailed calculations are performed for dose rates.

12.4 is basically the proof of the pudding, did we really adequately address occupational dose. We have calculated an annual occupational dose, including -- including an average of 50 person-rem.

MEMBER RYAN: Can you give me some idea of the average in the high per individual?

MR. PEREZ: Yes. I can get you --

MEMBER RYAN: Person rem are okay, but they really don't tell you what's cooking.

MR. PEREZ: Reactor operations and surveillance is 12 percent. Routine maintenance, 15 percent. 15 percent. In-service inspection, 17 percent. Special maintenance activity, 29 percent. Waste processing, 10 percent. Refueling, 16 percent.

I hope these add up to 100. I didn't check.

MEMBER RYAN: Well, that's -- yes,

that's interesting but, I mean, I'm interested in an individual, not -- I get -- you know, there are dose rate per person ranges from zero to 400 milligram a year.

MR. PEREZ: I can provide you that. The average dose rate, okay, the average dose rate in millirem per hour --

MEMBER RYAN: No, no, no. I'm interested in the annual dose per person.

MR. PEREZ: Per person. Okay.

MEMBER RYAN: Worker X gets 3 rem a year and worker Y gets .01 rem a year.

MR. PEREZ: I have that information divided by activities. That's something that I'll have to --

MEMBER RYAN: Actually, maybe do a take-home on that one, too.

MR. PEREZ: Okay.

MEMBER RYAN: To get a table, because I think it's helpful to understand the breakdown. It's nice to meet the metric or to evaluate against the metric, you know, as you've done in the slide.

MR. PEREZ: Right.

MEMBER RYAN: But it's really helpful

to understand the breakdown. I mean, where is the action? Is it -- it sounds like special activities and maintenance, you know, might be a big chunk of the total.

MR. PEREZ: Yes.

MEMBER RYAN: For example.

MR. PEREZ: Yes.

MEMBER RYAN: And so where does the -- where do you focus your activities then in radiation protection improvements, it's in those areas, so I'm just trying to get some insights there.

MR. PEREZ: Okay.

MS. PEDERSON: Okay. So, to reiterate your question, what you're looking for is the type of activities that annual dose per person?

MEMBER RYAN: Yes. And rather than having you read out all the numbers to me and have me try and scroll them down, hardly, if you could just continue on with the table and it gives me those worker categories to percent of the totals, and then by even job title, the average per year.

Like, you know, a maintenance worker is typically 1.3 rem per year. And then the per year doses by worker categories would be real helpful.

MR. PEREZ: And you want me to include an outage year?

MEMBER RYAN: Absolutely.

MR. PEREZ: Okay.

MEMBER RYAN: Yes. Yes, outage and typical would be great.

MR. PEREZ: A running three-year --

MEMBER RYAN: Perfect.

MR. PEREZ: -- including an outage.

MEMBER RYAN: Perfect.

MR. PEREZ: Okay.

MEMBER RYAN: Perfect. Yes, that's perfect.

MR. PEREZ: Yes, I understand. Yes, I'm sorry, that information is here --

MEMBER RYAN: Yes.

MR. PEREZ: But it will take -- it will take some time to just getting that form.

MEMBER RYAN: As a take-home, that's fine, and I think -- I think that gives us, you know, some insights as to, you know, how these designs are working and what's going on and so forth. That's helpful.

CHAIRMAN POWERS: And just set it up

so that Sandra actually has to present it.

MEMBER RYAN: There you go.

CHAIRMAN POWERS: Thank you, Pedro, sorry for the --

MEMBER RYAN: We want to make sure that she's on top of this.

MS. PEDERSON: Okay.

MEMBER STETKAR: Do you have any historical records from operations experience in France or Germany? I recognize the EPR is -- is somewhat different and you're trying to design a lot of this from the ground-up, but -- but there are several operating plants over there that are highly compartmentalized with a lot of concrete and many of the same design features.

Do you have any dose records from operating experience to give us a feel for historically how at least those plants have done?

MR. PEREZ: Yes. In fact, what happens here is the -- you know, the EPR and certain systems within the EPR resemble operating plants in Germany. So, the operating doses, the time to service the systems, we took from those plants.

MEMBER STETKAR: Okay.

MR. PEREZ: Okay. Then we adjusted dose rates because of our source term.

MEMBER STETKAR: Okay.

MR. PEREZ: Okay. So, the calculation we did --

MEMBER STETKAR: So you actually took the exposure times from --

MR. PEREZ: Yes.

MEMBER STETKAR: -- from their operating experience?

MR. PEREZ: Yes. And other experience was maybe French, depending on the system.

MEMBER STETKAR: Yes.

MR. PEREZ: Give an example, aeroball system. It doesn't exist in this country. So we looked at the German -- what do they do when they dismantle the reactor vessel, take out the -- you know, the aeroball system, how long does it take, these are activated components. So, yes, we absolutely had to. It was necessary.

MEMBER STETKAR: Well, that's the exposure times. I was curious in terms of actual accumulated dose, operating experience from those plants, recognizing that there are differences, but --

MR. PEREZ: It's all -- it's one composite document. It's in there, yes.

MEMBER STETKAR: Okay.

MR. PEREZ: It is in there. The same question --

MEMBER STETKAR: Sure.

MR. PEREZ: -- of, you know what was the, you know, the German worker dose rate, that's also available. You just have to manipulate the numbers.

But, yes, I definitely not only did I personally talk to these people, but we had the system engineers spend time in Germany, if the system was German, or be in France if the system is, you know, in France.

MEMBER STETKAR: Okay.

MR. PEREZ: And again, something that's very -- the industry data, again, comes from the NRC publication O-0173, Volume 29, I believe is what was most recently available, available publicly.

In my final slide is Section 12.1 talks about programs. And programs are for the COL applicant to fully describe in their application, keeping in mind as NEI has templates that are approved by the staff, we change our design certification to say the NEI template is an acceptable option to comply with this COLA item.

The latest has been NEI 08-08 that the staff approved relating to 10 CFR 20.1406. So we try to monitor and we update the classification as these templates become available.

We just received an email that -- I'll read it. Level 1 coatings are epoxy-based on an ASTMD standard 5144, described in 6.1.2 of the FSAR. And in Table 6.1-2.

MS. PEDERSON: It's ASTMD 5144 -- or I'm sorry, 5744, not 74.

MEMBER STETKAR: That is an ASTM standard.

MS. PEDERSON: 5744.

MEMBER STETKAR: Epoxy.

MR. PEREZ: That concludes my presentation, so I would be happy to answer --

MEMBER RYAN: Thank you for letting us ask questions as we go along.

CHAIRMAN POWERS: When are we scheduled to hear the article on radiation protection? Do you remember.

MR. WIDMAYER: I don't remember.

Chapter 12 is R-COLA. I have to check.

CHAIRMAN POWERS: Okay. They are going to have a fairly elaborate program description in there and it would be useful to look at those if it's, you know, contemporaneously available. I'm sure Mike would like to plough though it in great detail.

MEMBER RYAN: You betcha.

CHAIRMAN POWERS: Yes. I mean, it just make sense to look at that.

MEMBER RYAN: Maybe while we've got a couple of minutes, I'll just ask one sort of summing question, and it may be a question for NRC.

How's the translation process going from US Codes to French Codes to calculational methods back and forth? Has that been a challenge or --

MR. PEREZ: No. It depends. It was interesting early on in the process. I saw a document that had curies per cubic inch.

MEMBER RYAN: That's helpful.

MR. PEREZ: But we did just in time training, so the more spent -- the more time we spent it has been now -- in my area, I can speak sieverts or rem.

MEMBER RYAN: That's the easy part, but when you have a code you're trying to vet, I mean, using a US code that's understood by the NRC staff while you're importing French codes and they have to learn them, too, or how's that working?

MR. PEREZ: Well, radiation protection --

MEMBER RYAN: Yes.

MR. PEREZ: -- there was only one code used that is widely used in Europe, that's RANKERN.

MEMBER RYAN: Yes.

MR. PEREZ: The staff audit us because we used this non-US standard code called RANKERN.

MEMBER RYAN: Right.

MR. PEREZ: And the staff didn't talk about how they performed --

MEMBER RYAN: Okay. And we'll get to that when we talk to the staff. That would be great. Thank you.

Pedro, thank you for an excellent hour and a half or so. It was a great presentation.

CHAIRMAN POWERS: That was superlative.

MEMBER RYAN: It was a great presentation.

MR. PEREZ: My pleasure. Thank you.

MS. PEDERSON: Thank him.

MEMBER RYAN: Give him a gold star.

MS. PEDERSON: Thank you. I will. Thank you.

MR. PEREZ: And I just want to confirm, I think I had two take-aways, but I think the answer was --

CHAIRMAN POWERS: No, no, no. Sandra has two take-aways. Well, thank you again.

MS. PEDERSON: Thank you.

CHAIRMAN POWERS: We will recess for 15 minutes?

MEMBER RYAN: Yes.

CHAIRMAN POWERS: And come back and talk to the staff.

MS. PEDERSON: 2:30.

(Whereupon, the above-entitled matter went off the record at 2:15 p.m. and resumed at 2:31 p.m.)

CHAIRMAN POWERS: We are back in session, then.

MR. TESFAYE: Yes. Jason Jennings is

the Chapter project manager, Chapter 12 project manager, and he knows the ground rules and he will start by telling his bio.

CHAIRMAN POWERS: Have at it, Jason.

MR. JENNINGS: Good afternoon. My name is Jason Jennings, I'm the project manager for Chapter 12, which Getachew mentioned. I'm also the project manager for Chapter 11, Section 14.3 on ITAAC.

CHAIRMAN POWERS: You've got to quit -- you know, when they're passing out straws you've got to quit drawing the short one.

MR. JENNINGS: Because I also earned Chapter 2 for Bell Bend application.

A little bit on my background, I have about 17 years of nuclear experience, most of it in the operations end, which includes eight years enlisted in Nuclear Navy. I earned my Bachelor's degree from Thomas Edison State College in Nuclear Engineering Technologies while I was on active duty.

After the Navy I spent about five years at Oyster Creek. I was a licensed senior reactor operator there, qualified ship technical assistant and also a qualified site shift manager.

I've been with the NRC for the last four years. I started here as a member of the construction inspection program, development team when there was all of five of us working on that in NRR, and then later came over to NRO in that same position. I held that job until April of this past year when I joined NARP.

So, moving on to talk a little bit about Chapter 12, before we get into the presentation, I want to draw your attention to the last of the slide packages which you have up there. It does have some acronyms in it that we'll be referencing in the course of the presentation.

CHAIRMAN POWERS: But you don't have -- oh, you do have information on RAI. This one's defined differently than the other one. This is going to confuse.

MEMBER POWERS: I am.

CHAIRMAN POWERS: Oh, okay.

MEMBER POWERS: We knew that.

MR. TESFAYE: We just made this correction after we learned --

CHAIRMAN POWERS: I knew that. This is a learning organization here. I can tell.

MR. JENNINGS: The things you learn over lunch. Let's see, where was I?

Moving on with the slides, I can see from the table here there are a total of ten open items from the safety evaluation for Chapter 12, most of which fall under Sections 12.3-12.4, which are both covered in one -- one group.

Sara will provide some more detail on a few of the open items in the course of her presentation.

I think it's important to note that among these ten RAI's we don't see any major issues with them at this point. I don't think it's a show-stopper by any means.

CHAIRMAN POWERS: I think -- I believe we had a definition of terms last time that if we had show-stoppers we probably wouldn't be here.

MR. JENNINGS: Yes. On the next two slides I'm not going to read these to you here. There are some brief descriptions of the open items that's continued -- included in our package for completeness here.

So, at this point, I'll turn the presentation over to Sara Bernal of the Health Physics Branch in the Division of Construction, Inspection and Operational Programs.

MS. BERNAL: Good afternoon. My name is Sara Bernal. I am a health physicist in the Office of New Reactors. To tell a little bit about myself, I have a Bachelor's in mechanical engineering from the University of Michigan.

I also have a Master's in health physics from the University of Michigan. I've been working at the NRC for four years, starting in NRR, the Office of Nuclear Reactor Regulation, and then moving to the Office of New Reactors when that office was created.

I've completed the Nuclear Safety Professional Development Program, and I've also completed the Office of New Reactors Technical Reviewer Qualification Program.

CHAIRMAN POWERS: And you've taken the course on the New Reactors Safety --

MS. BERNAL: Yes, I have, and you were teaching it.

CHAIRMAN POWERS: And I take it that you enjoyed it immensely.

MS. BERNAL: Of course.

MEMBER STETKAR: Mission accomplished.

CHAIRMAN POWERS: We search these

things out whenever we can.

MEMBER STETKAR: Out of how many students you got one? This is good. And even that only under threats.

Go ahead, Sara.

MS. BERNAL: Okay. As the Applicant has stated, Chapter 12 of the application describes facility and equipment design features and programs which are used to meet the occupational range and protection standards of Part 20, Part 50, Part 52 and Part 70.

The rest of this presentation will highlight the most significant issues covered in my review.

For Section 12.1, the staff reviewed ALARA considerations which the Applicant applied to the design process, as well as to the equipment and facility design. AREVA discussed these in detail before this, but some examples, some highlights include training of AREVA design engineers on ALARA issues, lessons learned and regulatory guidance.

Also the use of 3-D computer modeling for driving piping diagrams to avoid streaming and avoid improper piping slope, ensure proper segregation of radioactive and nonradioactive piping, and reduced the use of fuel burn piping.

Also discussed was the use of low-cobalt alloys for key primary site components and the use of installed platforms where workers have historically had to install scaffolding, and also the compartmentalization which is used to a great degree in the EPR design.

CHAIRMAN POWERS: Let me ask you a question about the cobalt alloys. I mean, after the disasters and stellate and things like that, do we really have anybody proposing to use any cobalt alloys in those plants anymore? I mean, is that a problem, or -- I'm just curious.

MS. BERNAL: I don't think anybody's proposing to use --

CHAIRMAN POWERS: I mean, cobalt contamination nowadays is just the fact that you can't get nickel that's completely cobalt-free. I mean, that's it. I was just curious if anybody was actually --

MEMBER RYAN: Except in the waste generation with high cobalt stellate's gone way down.

CHAIRMAN POWERS: Yes, I know. I

mean, we went to the stellate disasters and then -- and like that's a lesson that's probably carved in some people's skulls some place.

I was just curious. Go ahead.

MS. BERNAL: Sure. In addition to a large design, 10 CFR 20.1101 also requires the licensees incorporate ALARA concepts into their procedures, and obviously this is an operational concern, and so it's outside of scope for the design certifications so the Applicant included a COL information item to have the COL applicants describe their ALARA operational program, and this is COL information item 12.1-1.

Section 12.2 of the application, the staff reviewed the applicant's description of contained and airborne radioactivity sources that were used as inputs for the shielding and ventilation designs.

In their evaluation the staff requested information on source strength for the spent fuel source, the safety injection system and the aeroball system such that staff could evaluate the EPR shielding design as well as access controls.

The incorporation of this source term

information into the FSAR is an open item that is being tracked.

CHAIRMAN POWERS: Did the staff just review shielding calculations, or did they do independent calculations?

MS. BERNAL: We did independent calculations also. We're in the process.

Section 12.2 also describes airborne sources for the EPR design. The staff was not able to reproduce the airborne concentrations using the details describes in FSAR, so we issued an RAI 280, Question 12.02-6, asking for greater detail in the applicant's methodology.

This question is being tracked as an open item.

Finally, Reg Guide 1.206 states that all COL applicants should describe any site-specific byproduct source or special nuclear materials that exceed 100 millicuries. The applicant has designated this as COL information item 12.2-1.

Section 12.3-12.4, radiation protection design features. For this section the staff reviewed the EPR facility and equipment design features for maintaining personnel exposures ALARA.

Some of these features were discussed in detail by the applicant and mentioned in a previous slide. During the review it was noted that there was an accessible portion of the spent fuel transfer tube, also previously mentioned by AREVA.

In response to an RAI, the applicant added gates to the design that would restrict access to this unshielded portion of the tube and this, combined with the labyrinth shielding prevents worker overexposures and therefore this design change is acceptable.

The staff also found that the application did not contain information describing design features which would prevent inadvertent reactor cavity drain down during refueling.

The staff issues RAI 280, Question 12.3-12.4-17, requesting that the applicant describe design features which would prevent inadvertent drain down, and the associated potential for work exposures from unshielded or poorly-shielded spent fuel.

This issue is being tracked as an open item. With respect to plant shielding design, the applicant performed a shield analysis to determine the radiation zones throughout the plant and to ensure adequate shield thicknesses.

In evaluating the analysis the staff is performing independent shielding calculations for select areas. These independent calculations also are -- reproducing calculations that were done using RANKERN, which is a British code that is not used in the United States, therefore we are having a contractor reproduce those calculations, using a code that is used in the United States, and NRC is familiar with.

CHAIRMAN POWERS: It's just the unfamiliarity. It's not -- there's nothing of which you've identified deficient about RANKERN?

MS. BERNAL: Right. There's just no knowledge about it.

MEMBER RYAN: What is the point there? I guess you're going to do side-by-side calculations and if they all match up over a range of values and inputs you're going to declare victory or --

MS. BERNAL: Right.

MEMBER RYAN: -- you concur with the calculations? And if there's some deviation you'll figure out what it is and go from there?

MS. BERNAL: Right.

MEMBER RYAN: Do you plan on writing a report about that or how will you document that, you know, whatever the outcome is?

MS. BERNAL: We'll document the results of whether analyses show that the -- confirm the applicant's application information, and then we also document -- we have a contractor document, their analysis in a report that they provide us.

MEMBER RYAN: It might be helpful to just put a place over that that might be something the subcommittee might like to hear about when that's captured.

MS. BERNAL: Sure.

MR. TESFAYE: That's the staff -- that's the staff-controlled reactors. I assume the final SE would include some summary of that report.

MEMBER RYAN: Right. Right. But I think you had some of the details of that --

MR. TESFAYE: Yes.

MEMBER RYAN: -- at the appropriate meeting when it's timely would be helpful.

MR. COLACCINO: This is Joe Colaccino. When we discussed that here, we confirmed that, that that -- that this is a confirmatory item in the NRC right now, so my assumption would be is that we would document that in the safety evaluation so that the next time that we meet on Chapter 12 and we discuss the closure of that, we can also go over and take a note to go over the closure of the documentation, the confirmatory analysis.

MEMBER RYAN: Great. Thank you.

MR. COLACCINO: Thank you.

MS. BERNAL: And the ventilation design was described earlier by AREVA, by Pedro, it reduces occupational exposures by maintaining pressure differences so that air flows from areas of low potential for contamination to areas of higher potential for contamination.

Ventilation also, of course, draws airborne radioactivity from the spent fuel pool area to keep worker exposures ALARA.

In addition, in Section 12.3-12.4, the staff reviewed the location and design criteria applied to the applicant's area and airborne monitoring system.

The staff found this criteria to be acceptable except for the issue of calibration. There's -- we issued an RAI 295, Question 12.03-12.04-18, asking the applicant to provide a description of the calibration methodology so that we could assess compliance with 10 CFR 20.1501-B, still being tracked as an open item.

The applicant also performed an analysis to demonstrate that operators could assess or access radiological vital areas within the facility and not exceed five rem whole-body dose in accordance with 10 CFR 50.34F27, and the criteria of NUREG 07372B2.

Staff asked the applicant to revise the calculated operator doses to include the contribution from airborne radioactivity. The applicant provided revised mission doses which are acceptable.

Just a note on there, we also reviewed parts of Chapter 7, 9 and 11 for details on the accident monitors and ventilation design, and details on the airborne radiation monitors.

The applicant also performed a dose assessment that resulted in projected annual exposure of 50 person rem. This assessment, as discussed, was based on current reactor operating experience in the EPR's ALARA design considerations.

50 person rem compares favorably to the 2007 three-year average annual PWR exposure of 78 person rem.

Finally, the staff reviewed the application for compliance with 10 CFR 20.1406. 20.1406 requires that the applicant describe how the facility design minimizes contamination of the facility and the environment, as well as facilitates eventual decommissioning, and minimizes radioactive waste generation.

In Chapter 12, the applicant describes their general approach to compliance and also provides specific facility equipment design features. These features were reviewed by myself as well as other technical branches.

The EPR 20.1406 philosophy focuses on prevention and containment of leaks, and towards that end most tanks are located indoors inside buildings.

The one exception to this is the demineralized water distribution system which has two tanks located out of doors and which is connected to several radioactivity-containing systems, systems which contain radioactive fluids.

The staff asked RAI 228, Question 12.3-12.4-10, requesting the applicant to describe how this design prevents contamination of the two tanks located outside as well as whether the system has any buried piping, and how that -- and if so, how the buried piping can be monitored for leakage or inspected for leakage.

This is being tracked as an open item. 20-1406 also has an operational component to compliance. Operational procedures are outside the scope of the design cert, so the staff asked in RAI, RAI 23, Question 12.03-12.04-1, asking the applicant to address compliance with the operational requirements incorporating the COL information item into the FSAR would address this issue, would address this open item.

Next slide. There are three COL information items in Section 12.3-12.4. These are summarized on the slide. Briefly, the COL applicant will provide site-specific information on sampling, recording and reporting of airborne releases of radioactivity.

The applicants are also -- the

information item will also ask the applicant to provide estimated annual doses to construction workers from existing operating plants.

And the third item there is a request stating that applicants will provide the use of -- or describe the use of portable instrumentation to demonstrate compliance with 10 CFR 50.34(f)(2)-xxvii), and the criteria in item 3D.33 of NUREG-0737.

Finally, the last section, Section 12.5. Section 12.5 of the application addresses the operational radiation protection program which is out of scope as far as the design certification application, therefore it consists of one COL information item.

The COL information item states that the COL applicant will fully describe the operational radiation protection program, including the organization, equipment, instrumentation and facilities and procedures.

And that's the end of my presentation, if you have any questions.

CHAIRMAN POWERS: Any questions on this?

MEMBER RYAN: Just a couple. Could

you talk about what we touched earlier on in Pedro's presentation about how you go about verifying the calculations you got from the applicant? Could you talk a little bit more about that, when you did the same code and same calculation, a different code?

You mentioned one case. What else have you done in that area? I didn't know what you got makes sense or is right or --

MS. BERNAL: Well, we have Oak Ridge National Lab using the inputs that AREVA used as input to the RANKERN calculations, so they understand the source term, the geometry of the source, and they use that as inputs to a code that they're familiar with and that they understand the underlying physics.

And then they document whether the results are the same as what RANKERN has put out. RANKERN has --

MEMBER RYAN: Do you have any other examples where you've done that sort of confirmatory calculation at this point?

MS. BERNAL: Well, RANKERN is the only code they used that was not a US code, but we've also done calculations to -- around areas where there are high dose rates to verify what the applicant's dose rates are.

For example, around the spent fuel transfer tube, we've done calculations to see what -- whether the shielding is adequate, what kind of results, dose rates, what are the radiation zones, are they -- are they correct.

We did the spent fuel transfer tube. We're also doing significant source in the other buildings where there are high dose rates and potential for worker overexposure, for example, the volume control tank inside the fuel building.

MR. ROACH: This is Ed Roach. I'm acting branch chief for Health Physics, New Reactors, and I'm also the technical monitor for the contract we have with Oak Ridge National Labs for the dose rate calculations, source term calculations for the new reactor, DCD's.

And as Sara stated, the work we did was selectively go though the design of the plant, look at the most, what we considered the most risky areas, the highest dose rates, and areas that would not lend themselves to a simple bicrucial calculation.

And then when we found out, asked

questions about that, Sara did, and we found a code that we were not familiar with, was not in the RSIC inventory at Oak Ridge, and so we decided to initiate selective source term determination so they would determine the source term and do the dose rate calculation to the nearby zones that we were concerned with.

And Oak Ridge presents a report to us that we review, and then we will incorporate the results into our safety evaluation.

MEMBER RYAN: Great. Thanks. Okay.

MR. TESFAYE: If there are no questions, I think that concludes the staff's presentation of Chapter 12.

CHAIRMAN POWERS: Well, thank you.

MR. TESFAYE: And we appreciate you let us -- letting us present the staffers' Chapter size A, 2, 12 and 10 in the same group. We really appreciate that flexibility.

CHAIRMAN POWERS: I mean, I think we can tolerate it. It's --

MR. TESFAYE: We'll try to improve on it.

CHAIRMAN POWERS: To be honest with you, I got interrogated very hard by Commissioner

Klein on why we were doing that, and he said this doesn't -- this didn't strike him as a good idea, and I said if we were having troubles I would report back to him, but up till now we were having no troubles and I lauded the quality of both the staff -- I was disgusting.

I said both staff and licensing were doing such a wonderful job that it was just not posing a problem to us. But, you know, people are -- they're aware this is happening and they are asking us if we can live with it and so far we're just not having a problem doing it.

And that may be more difficult as we get into some of the other areas for, you know, some chapters, especially as we get to 15 in the PRA, we're liable to have rougher sledding there, but I think the quality of the presentations and the quality of analysis maintained on the same high level that we've had up till now, we'll sail through this. And the only person that's suffering is Sandra. Right?

MS. SLOAN: That's right.

CHAIRMAN POWERS: Any questions?

MR. WIDMAYER: Dr. Powers, to answer your question from earlier, we are scheduled to do

Chapter 12 of the R-COL at a meeting in April of 2010.

CHAIRMAN POWERS: That's close enough.

MR. WIDMAYER: Is that close enough?

CHAIRMAN POWERS: Yes. Yes. I've got no problems with that. I mean, it falls on your shoulders most heavily, but -- I don't think that's -- I mean, if it was 2011, then I would say, darned, and that's about all I would say, but -- I mean, that's contemporaneous enough so that we could -- we will retain some from one to the next.

Any other questions for this? As I say, we are not going to -- there will be no summary of this given at the full committee meeting in December. I undoubtedly will say something, that we had the meeting. I will probably say something really snotty about Sandra and -- I won't.

But, we will collect comments on the chapters probably sometime in February. and with that, I'll bring this meeting to an end. Thank you all.

(Whereupon, at 2:55 p.m., the meeting concluded.)

AREVA NP Inc.

Presentation to ACRS U.S. EPR Subcommittee

Design Certification Application

FSAR Tier 2 Chapter 10

Kevin Connell, Manager,
Mechanical/Plant Design Group

Chapter 10, Steam and Power Conversion System: Chapter Topics

- Summary description
- Turbine generator
- Main steam supply system
- Other features of steam and power conversion system

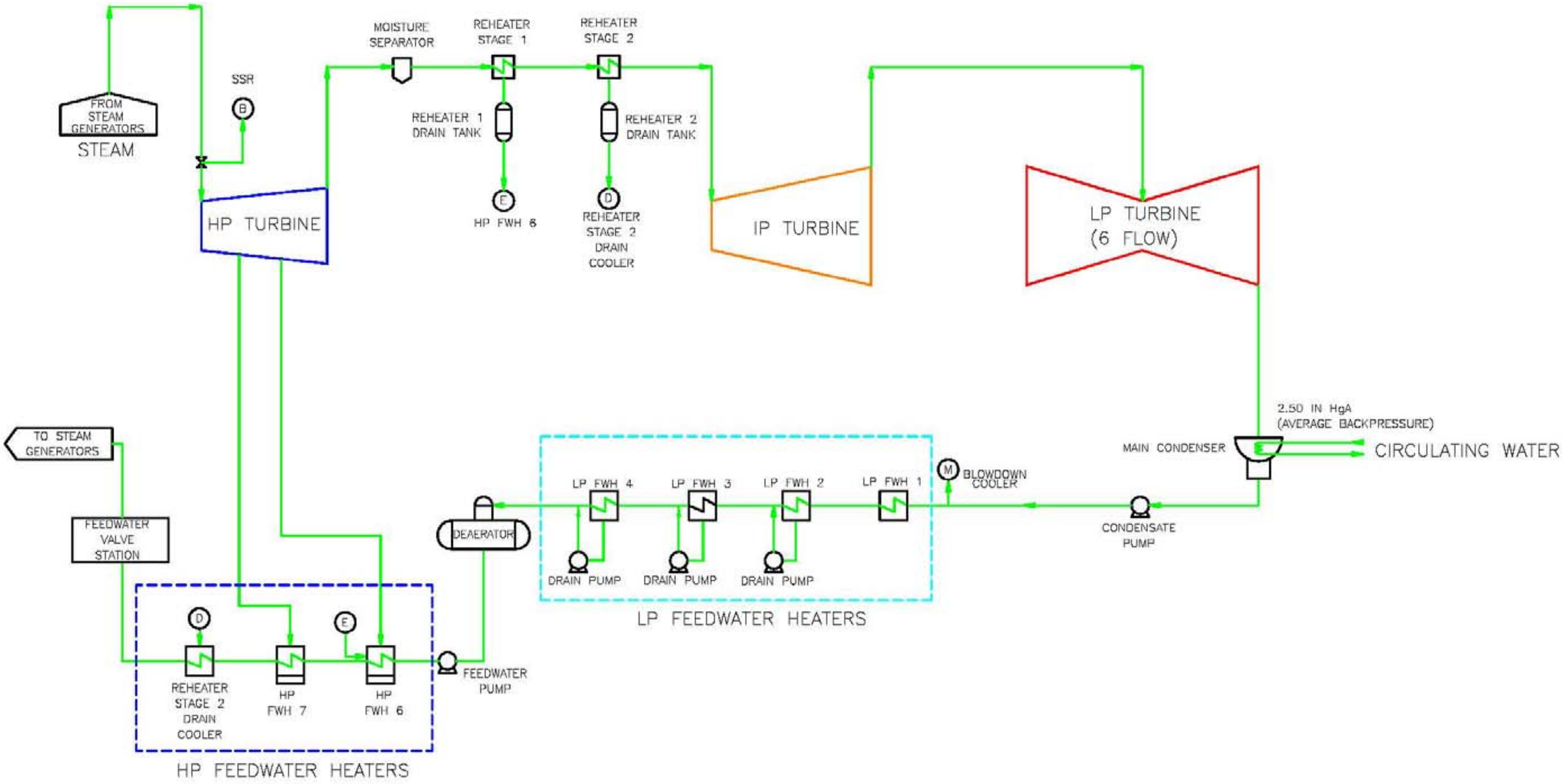
Chapter 10, Steam and Power Conversion System: 10.1 Summary Description

- **U.S. EPR™ design features that are fundamentally the same as previous designs**
  - **Turbine Generator**
  - **Main Steam System**
  - **Feedwater System**
  - **Condensate System**
  - **Turbine Gland Steam System**
  - **Condenser**
  - **Condenser Evacuation System**
  - **Circulating Water System**
  - **Steam Generator Blowdown System**

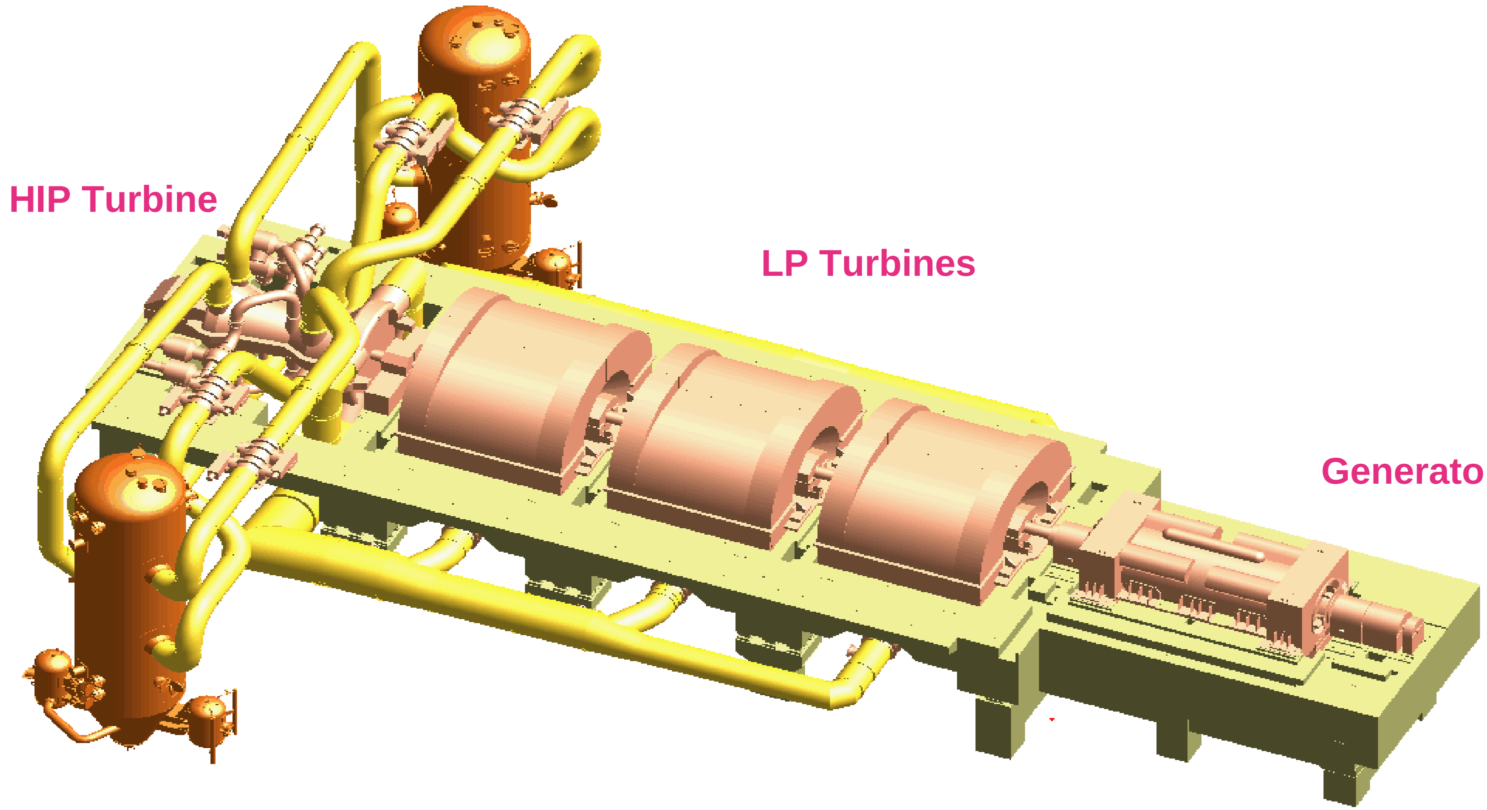
Chapter 10, Steam and Power Conversion System:
10.1 Summary Description

- **What is Different?**
  - **Single Flow High Pressure (HP) Turbine and Single Flow Intermediate Pressure (IP) Turbine in a common casing**
  - **Stand alone Startup/Shutdown Feedwater System - includes a smaller Startup/Shutdown Feedwater pump**
  - **All Emergency Feedwater pumps are motor driven**
  - **Two redundant and diverse electrical overspeed trip systems for the Turbine Generator**

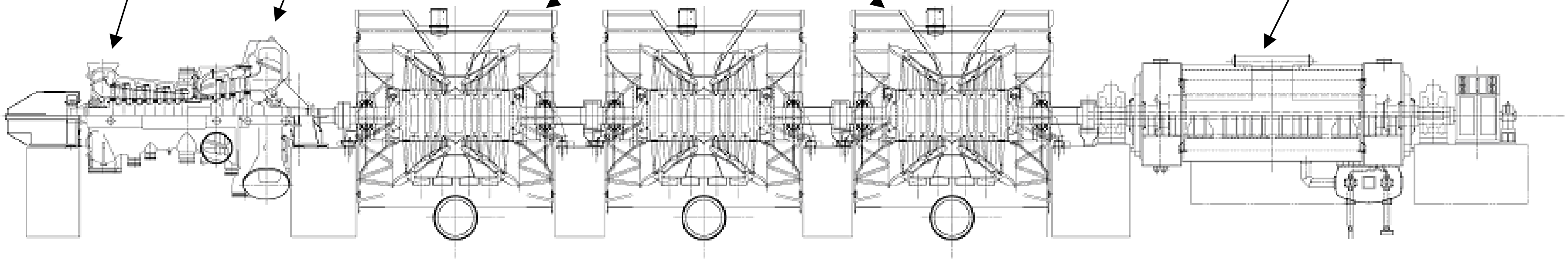
Chapter 10, Steam and Power Conversion System:
10.1 Summary Description



Chapter 10, Steam and Power Conversion System:
10.2 Turbine Generator



Chapter 10, Steam and Power Conversion System:
10.2 Turbine Generator

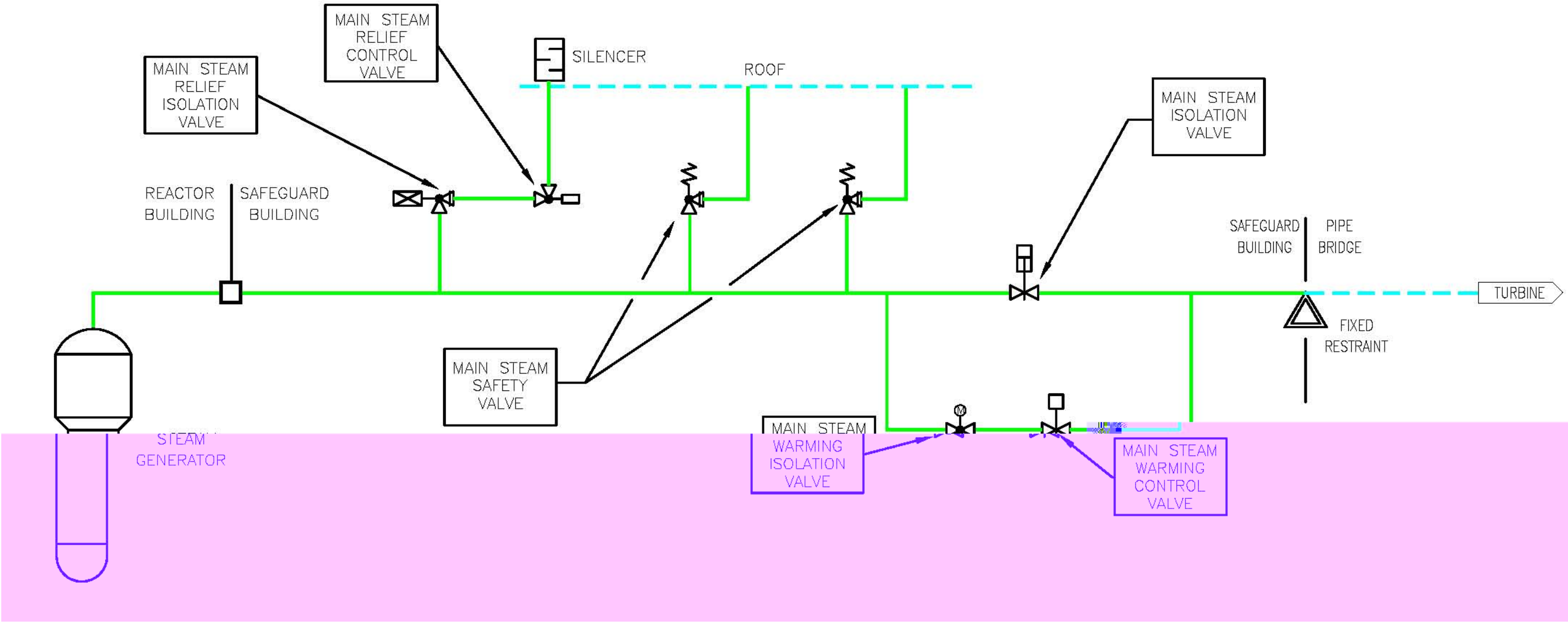


- **HP / IP / LP Sections**
- **Single flow HP section of HIP module receives steam through 4 steam leads, each lead with a stop valve and a control valve, utilizing full arc admission. Exhaust is to the MSR**
- **Single flow IP section of HIP module receives steam from hot reheat steam from MSR through 4 steam inlet pipes, each with a stop valve and an intercept valve. Exhaust is to the 3 LP turbines**
- **Double flow LP turbine receives steam from the IP turbine. Exhaust is to the main condenser**

Chapter 10, Steam and Power Conversion System:
10.2 Turbine Generator

- **Overspeed Protection**
  - **Two redundant and diverse electrical overspeed trip systems that meet single failure criterion**
  - **Overspeed trip systems are independent of the turbine governor system**
  - **Each system is designed and manufactured by a different vendor**
  - **Each system installed in separate cubicles with separate power supplies**
- **Turbine Rotor Integrity / Turbine Missiles**
  - **Rotor assembly is a series of welded forgings**
  - **Meets the turbine missile requirements of FSAR Section 3.5**
  - **Materials - vacuum melted/degassed Ni-Cr-Mo alloy**
  - **COL applicant will procure a turbine that meets or exceeds the FSAR bounding specifications, or provide suitable justification for the departure**

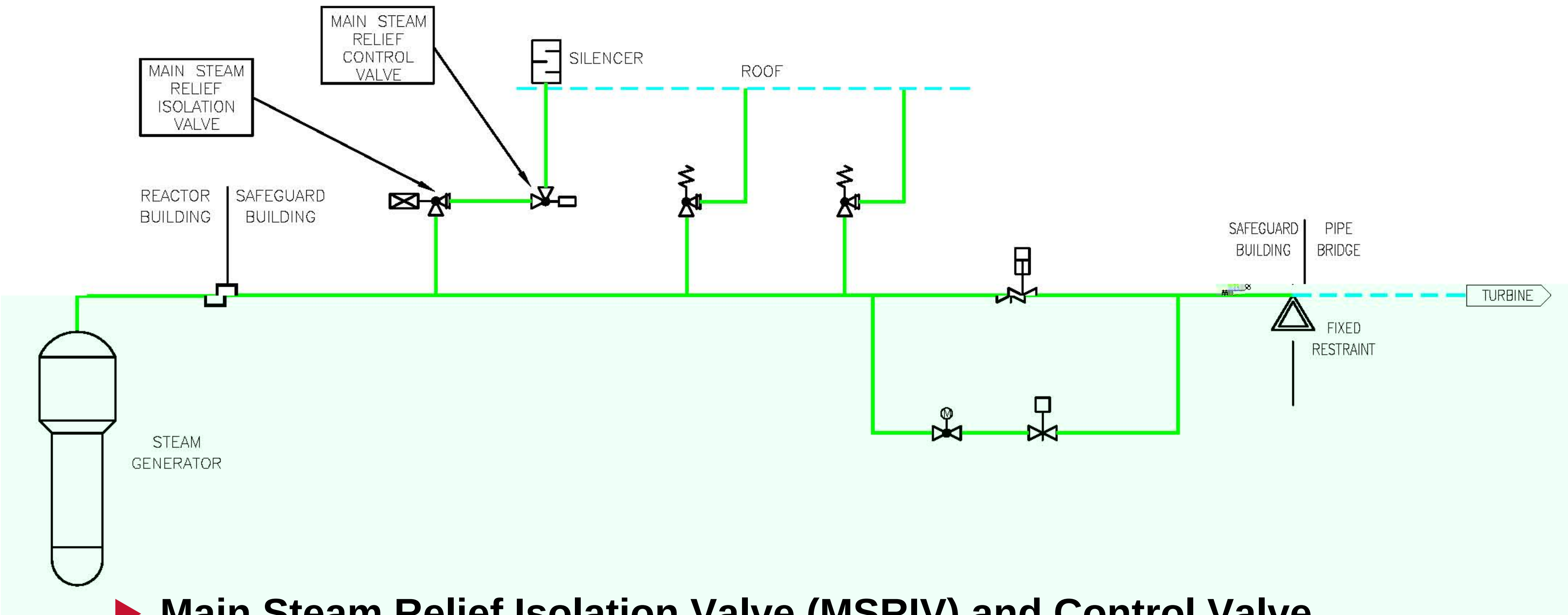
Chapter 10, Steam and Power Conversion System:
10.3 Main Steam Supply System



One of Four Divisions of Main Steam Supply

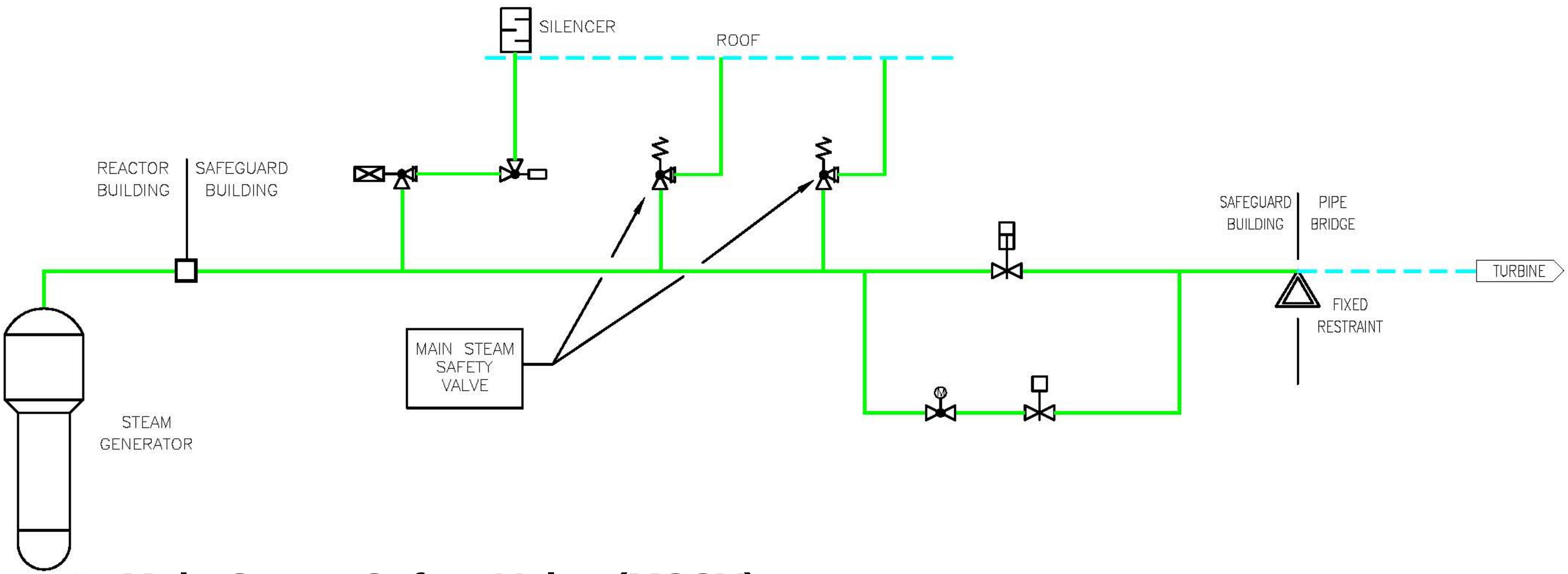
Chapter 10, Steam and Power Conversion System:

10.3 Main Steam Supply System



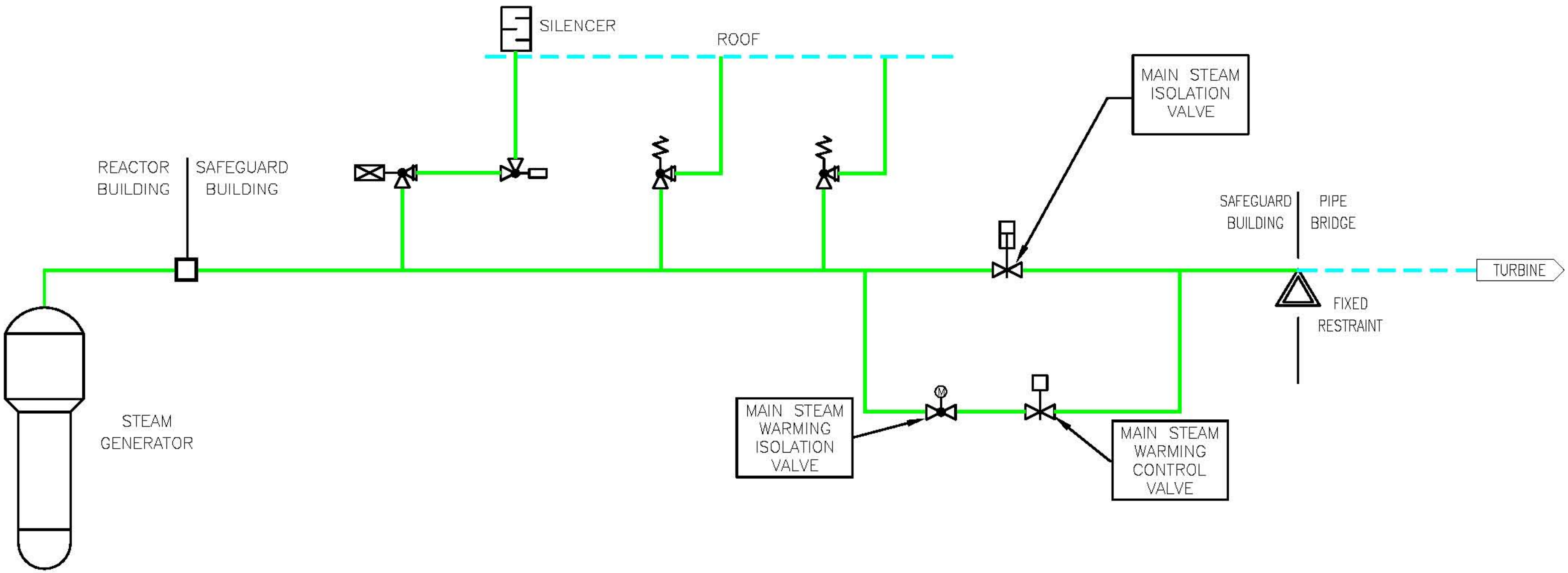
- **Main Steam Relief Isolation Valve (MSRIV) and Control Valve (MSRCV)**
  - **MSRIV / MSRCV capacity is 50% full power MSSS line flow (2.844 x 106 lbm/hr)**
  - **set pressure: 1,370 psig**

Chapter 10, Steam and Power Conversion System:
10.3 Main Steam Supply System



- **Main Steam Safety Valve (MSSV)**
  - **Two MSSV per relief train**
  - **Each MSSV capacity is 25% full power MSSS line flow (1.422 x 106 lbm/hr)**
  - **set pressure: 1,460 psig (1st MSSV) and 1,490 psig (2nd MSSV)**

Chapter 10, Steam and Power Conversion System:
10.3 Main Steam Supply System



- **Main Steam Isolation Valve (MSIV)**
  - **Closure in 5 seconds or less against 1320 psid and 5x(10)6 lbm/hr**
- **Main Steam Warming Isolation (MSWIV) and Control Valves (MSWCV)**

Chapter 10, Steam and Power Conversion System:
10.4 Other Features

- **Main condensers**
  - **Performs no safety function**
  - **Multi-Pressure, 3 shell deaerating condenser designed to HEI Standards**
  - **COL Applicant will select condenser materials dependent on site water source**
  - **Designed for turbine bypass flow of at least 50% of main steam flow without tripping turbine due to high backpressure**
  - **Hotwells are compartmentalized to help in identifying location of leakages**

Chapter 10, Steam and Power Conversion System:
10.4 Other Features

- **Main Condenser Evacuation System (MCES)**
  - **Performs no safety function**
  - **Two redundant holding vacuum pumps per condenser shell**
  - **One main hogging vacuum pump for startup**
  - **Exhaust from the vacuum pumps is routed to the turbine building air vent system**

Chapter 10, Steam and Power Conversion System:
10.4 Other Features

- **Turbine Gland Sealing System (TGSS)**
  - **Performs no safety function**
  - **Collects seal leakage from main steam stop and control valves, HP and IP turbine shaft seals.**
  - **Provides sealing steam to the LP turbine shaft seals.**
  - **Steam and non-condensables routed to the gland steam condenser**
  - **Air and non-condensables exhausted using two redundant exhaust fans**

Chapter 10, Steam and Power Conversion System:
10.4 Other Features

- **Turbine Bypass System (TBS)**
  - **Performs no safety function**
  - **Sufficient capacity to prevent actuation of main steam relief train following a turbine trip or full load rejection**
  - **Six turbine bypass valves but only five valves are required to bypass 50% of main steam flow**
  - **Normal operation:**
    - start up steam is bypassed to the condenser
    - shutdown cooldown rate regulated by TBS

Chapter 10, Steam and Power Conversion System: 10.4 Other Features

- **Circulating Water System (CWS)**
  - **Performs no safety function**
  - **Supplies cooling water to the main condenser and auxiliary cooling water system**
  - **Rejects heat to the environment via the normal heat sink**
  - **Design Certification scope limited to inside the turbine building**
  - **COL Applicant will:**
    - design site specific portion (outside Turbine Building)
    - select condenser materials dependent on site water source
- **Condensate Polishing System (CPS)**
  - **Performs no safety function.**
  - **Removes corrosion products and impurities from the condensate during startup to meet condensate and feedwater system water chemistry specifications (EPRI Water Standards)**
  - **Design capacity of 1/3 of full condensate flow**

Chapter 10, Steam and Power Conversion System:
10.4 Other Features

- **Condensate and Feedwater Systems (CFS)**
  - **Operational Functions:**
    - Provide water to the steam generators at the required temperature, pressure and flow rate.
    - Perform warm-up functions during startup and cool down functions during shutdown.
  - **Safety Functions:**
    - The CFS system provides redundant isolation of the condensate and feedwater systems for the following:
      - Prevent or reduce an overcooling event due to control malfunctions or line breaks.
      - Prevent depressurization of steam generators in the case of a feedwater line break.
      - Retain activity in the affected steam generator in the event of a steam generator tube rupture.
      - Shut off the feedwater supply to prevent containment overpressurization in the case of a main steam or main feedwater line break.

Chapter 10, Steam and Power Conversion System:
10.4 Other Features

- **Steam Generator Blowdown System (SGBS) –**
  - **Operational Functions:**
    - Removes the contaminants and minerals gathered by phase separation in the SG
    - Demineralized blowdown is returned to the steam and condensate systems.
    - Continuous monitoring of the blowdown water, with detection of steam generator tube leaks
    - Capability to discharge blowdown condensate to the Liquid Waste System upon detection of leakage to the secondary system
  - **Safety Functions:**
    - Isolation on EFW actuation
    - Provides containment isolation
    - Isolates the affected SG in case of SGTR

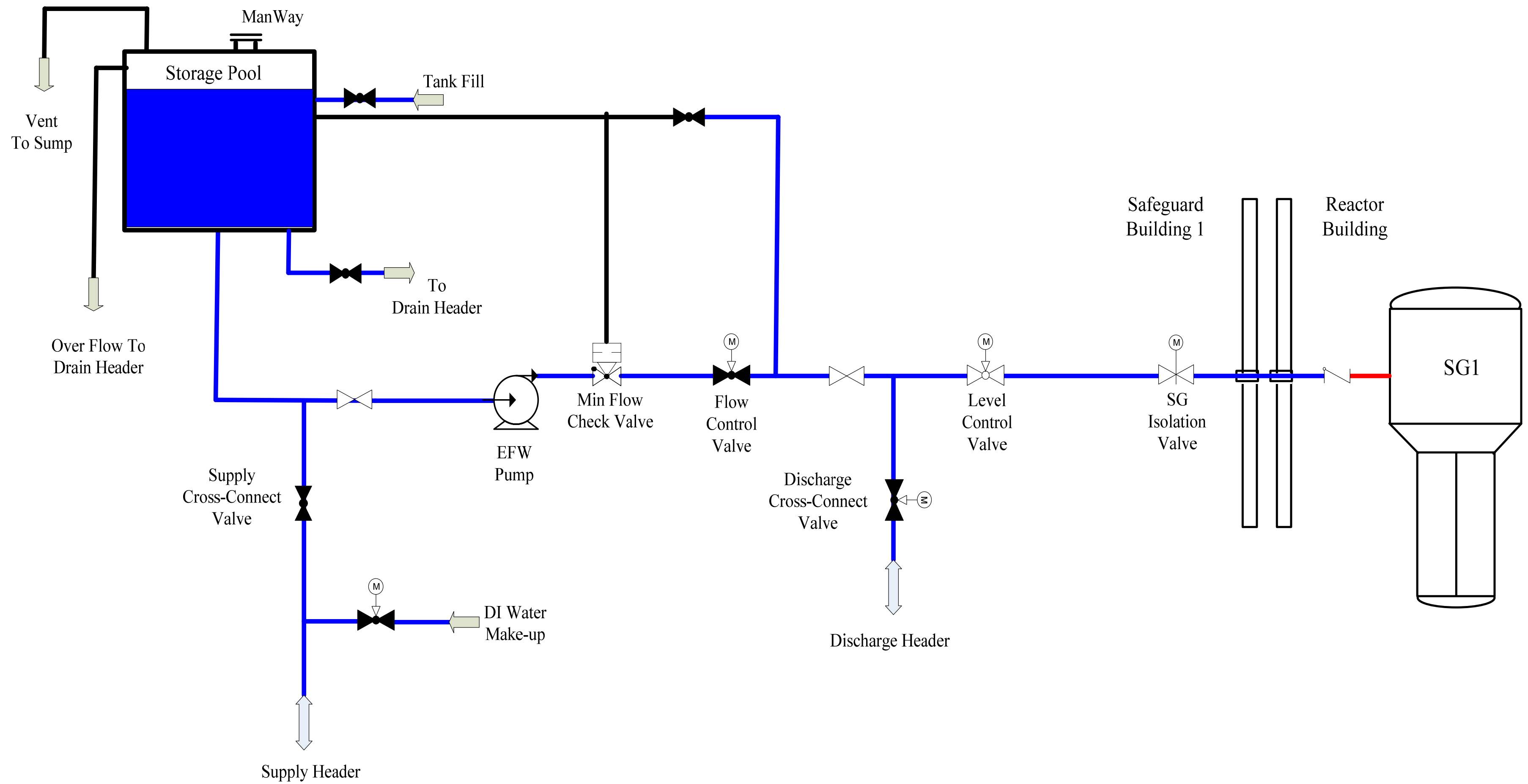
Chapter 10, Steam and Power Conversion System:
10.4.9 Emergency Feedwater System

- **Primary System Function:**
  - Provide flow to the SGs to restore and maintain SG water inventory
  - Support residual heat removal from the RCS
  - Cool down and depressurization of the RCS to RHRS entry conditions following design basis events
- **Four identical safety related trains each normally aligned to a separate Steam Generator**
- **Major components of each train:**
  - Storage pool
  - Centrifugal pump and motor
  - Flow control valve (FCV), level control valve (LCV), SG isolation valve (SGIV), and supply cross connect valve (SCCV) and discharge cross connect valve (DCCV)
  - Required instruments and controls

Chapter 10, Steam and Power Conversion System:

10.4.9 Emergency Feedwater System

Train 1 of 4



Chapter 10, Steam and Power Conversion System:
10.4.9 Emergency Feedwater System

- **Special Features**
  - **Four separate storage pools are provided, each enclosed within a separate Seismic Category I and tornado protected Safeguard Building**
    - Total available volume of the four pools is 411,200 gallons
    - Bounding cool down per BTP 5-4 requires 365,000 gallons
  - **Supply and discharge cross-connect headers provide the capability for each EFW pump to take suction from any storage pool and feed any SG**
  - **EFWS does not provide normal auxiliary feedwater functions**
    - Start-up and Shutdown System provides feedwater during plant start-up and shutdown and provides feedwater upon a loss of the Main Feedwater System

Chapter 10, Steam and Power Conversion System: 10.4.9 Emergency Feedwater System

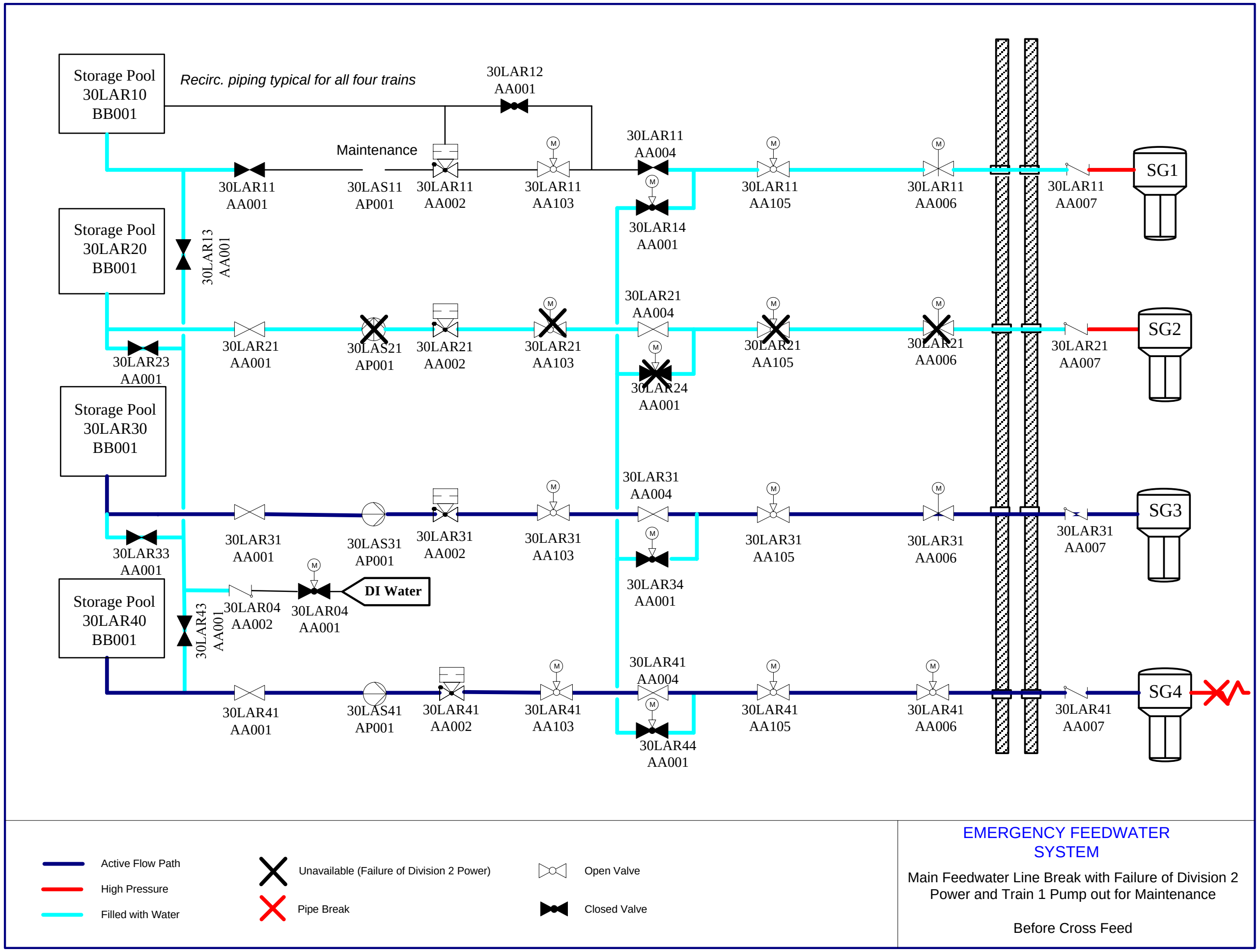
Electrical Power Supply

- **Each EFWS train is powered from a separate division of the Emergency Power Supply System (EPSS)**
  - A 6.9 kV switchgear is located in each Safeguard Building that powers one EFWS train pump and the associated 480V distribution panels that power the EFWS train motor operated valves.
  - In case of loss-of-offsite power, each of the 6.9 kV buses is powered by a separate Emergency Diesel Generator (EDG).
  - Essential valves (Level Control, Discharge Cross Connect, & Isolation) also fed from the Emergency Un-interruptible Power Supply (EUPS)
  - During EDG maintenance, alternate feed from the adjacent electrical division (1 to 2, 2 to 1, 3 to 4, or 4 to 3) will be provided to essential valves
  - In addition to the EPSS, the two diverse SBO diesels can supply power to essential EFWS equipment during a postulated station blackout event

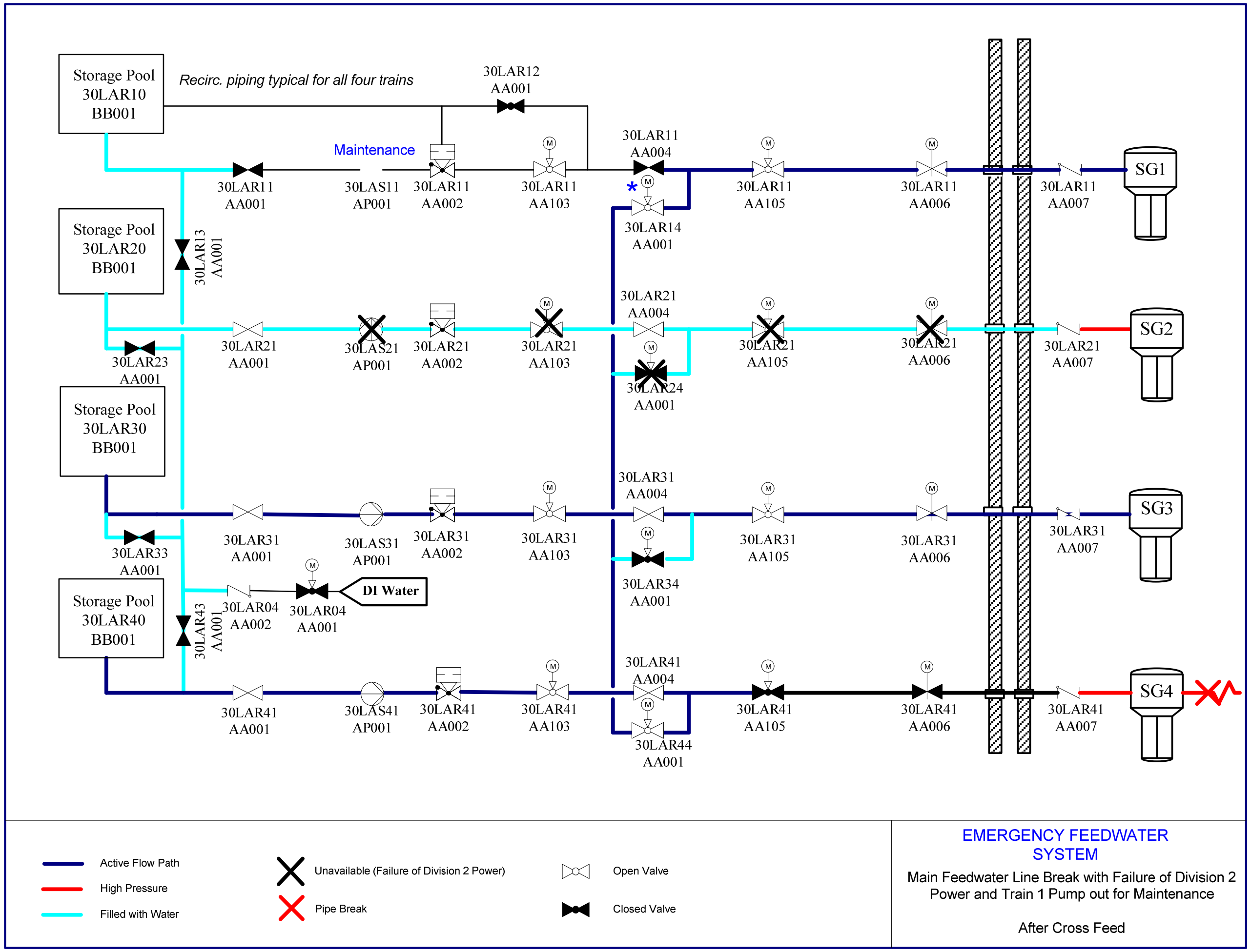
Chapter 10, Steam and Power Conversion System:

10.4.9 Emergency Feedwater System

Cross Feed Example



Chapter 10, Steam and Power Conversion System:
10.4.9 Emergency Feedwater System
Cross Feed Example



Chapter 10, Steam and Power Conversion System:
10.4.9 Emergency Feedwater System

- **Reliability and Diversity**
  - **Objective: Highly reliable EFWS that contributes to low core damage frequency**
  - **The EFWS achieves the reliability target requirements of 10CFR50.34(f)(1)(ii) through a combination of redundancy and diversity.**
  - **Turbine-driven pumps were not included due to:**
    - low reliability
    - extra maintenance requirements
    - associated high energy piping and environment / equipment qualification issues

Chapter 10, Steam and Power Conversion System: 10.4.9 Emergency Feedwater System Reliability and Diversity

- **A study of adding turbine driven pumps concluded:**
  - **Overall EFWS reliability is equivalent**

Initiating Event	Secondary Cooling Systems Credited	Probability that the credited systems fail to provide adequate SG Flow	
		4 Motor-Driven Pumps	2 Motor and 2 Turbine Driven Pumps
General Transient	MFW, SSS, and EFWS	3.55 E-07	4.00 E-07
Loss of Main Feedwater	EFWS	3.89 E-05	2.90 E-05
Loss of Main Feedwater	EFWS and SSS	1.14 E-05	8.35 E-06
Loss of Off-Site Power	EFWS (without power recovery)	1.01 E-04	1.37 E-04
Loss of Off-Site Power	EFWS (with off-site power recovery)	7.15 E-05	8.62 E-05
Loss of Off-Site Power	EFWS and SSS (with off-site power recovery)	5.68 E-05	7.50 E-05
Loss of Off-Site Power	EFWS (SBO conditions)	2.20 E-2	2.95 E-2

Chapter 10, Steam and Power Conversion System: Conclusion

- Steam and Power Conversion System removes energy from the RCS via the Steam Generators and converts it to electric power in the turbine-generator
- Provides safety-related heat removal via the emergency feedwater system and the main steam reliefs

Backup Supplemental Slides

10.1 Summary Description

10.0 Steam and Power Conversion System

This section describes the steam and power conversion system for the U.S. EPR.

10.1 Summary Description

The steam and power conversion system removes energy from the reactor coolant in the steam generators and converts it into electric power in the turbine-generator. The steam and power conversion system comprises the following process systems:

- Turbine-generator (TG).
- Main steam supply system (MSSS).
- Main condensers.
- Main condenser evacuation system (MCES).
- Turbine gland sealing system (TGSS).
- Turbine bypass system (TBS).
- Circulating water system (CWS).
- Condensate cleanup system.
- Condensate and feedwater system.
- Steam generator blowdown system (SGBS).
- Emergency feedwater system (EFWS).

The following portions of the steam and power conversion system have safety-related functions:

- Main steam piping between each SG outlet nozzle and the fixed point restraint downstream of its respective main steam isolation valve (MSIV) (refer to Section 10.3).
- Main feedwater piping between each SG inlet nozzle and the fixed restraints upstream of its respective feedwater isolation valves (refer to Section 10.4.7).
- Containment isolation valves for condensate system cooling water supply and return to the first stage SG blowdown cooler inside containment (refer to Section 10.4.7).
- SG blowdown piping between each SG blowdown nozzle and its respective outermost CIV (refer to Section 10.4.8).
- EFWS (refer to Section 10.4.9).

10.2 Turbine Generator

10.2 Turbine-Generator

This section describes the turbine-generator (TG) for the U.S. EPR.

The TG converts the thermal energy supplied by the main steam supply system (MSSS) into electrical energy.

The TG package interfaces with the MSSS at the high pressure (HP) turbine stop valves. It interfaces with the condensate and feedwater system at the low pressure (LP) turbine exhausts to the condenser, at the turbine extraction connections to the feedwater heaters, and at the moisture separator reheater (MSR) condensate and drain tank outlet connections to feedwater heaters and drain coolers. The generator terminals and enclosure connect to the isolated phase buses and ducts. The TG control system interfaces with the plant process automation system (PAS).

10.2.1 Design Bases

The TG performs no safety-related functions and therefore has no nuclear safety-related design bases.

The TG principal design features include:

- The TG is designed for base load operation. The design of the TG has provisions for load follow operation for future consideration.
- The TG is capable of a load step (increase or decrease) of 10 percent of rated load below a 50 percent power level or a ramp rate (increase or decrease) of 5 percent per minute of actual load when the power level is in the range of 50 to 100 percent, without necessitating a turbine trip.
- The TG is designed to trip automatically under abnormal conditions.
- The TG load change characteristics are compatible with the instrumentation and control system which coordinates TG and reactor operation.
- The TG is designed to accept a sudden loss of full load without exceeding design overspeed.
- The TG is designed to permit periodic testing of steam valves important to overspeed protection, emergency overspeed trip circuits and several other trip circuits under load.
- The failure of any single component does not cause the rotor speed to exceed the design speed.
- The reheat stop and intercept valves are capable of closure concurrent with the HP turbine stop valves, or of sequential closure within an appropriate time limit, to make sure that turbine overspeed is controlled within acceptable limits.

10.3 Main Steam Supply System

10.3 Main Steam Supply System

The main steam supply system (MSSS) conveys steam from the steam generators (SG) to the high pressure (HP) turbine. The MSSS also provides steam to the second stage reheaters, deaerator pegging steam and backup auxiliary steam.

10.3.1 Design Bases

The MSSS provides the following safety-related functions:

- Isolate main steam lines in the event of excessive steam flow to prevent over cooling of the reactor coolant.
- During accident conditions, provide initial residual heat removal by venting steam to the atmosphere via the main steam safety valves (MSSV) and main steam relief trains (MSRT).
- In the event of a steam generator tube rupture (SGTR), retain activity by steam side isolation.

The MSSS has the following design basis requirements and criteria:

- Safety-related portions of the MSSS are designed to withstand the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, tsunamis and seiches without loss of capability to perform its safety functions (GDC 2).
- Safety-related portions of the MSSS are designed to withstand the effects of external missiles and internally generated missiles, pipe whip and jet impingement forces associated with pipe breaks (GDC 4).
- Safety-related portions of the MSSS are not shared among nuclear power units (GDC 5).
- Safety-related portions of the MSSS are designed to maintain fuel and reactor coolant pressure boundary (RCPB) design limits by providing sufficient cooldown capacity and suitable power supply and redundancy to assure functionality during a loss of offsite power (LOOP) (GDC 34).
- Safety-related portions of the MSSS are designed to provide decay heat removal capability necessary for core cooling and safe shutdown during a station blackout (SBO) event (beyond design basis accident) (10 CFR 50.63).

The MSSS is designed to meet the following functional criteria:

- Convey steam from the SGs to the HP turbine and second stage reheaters at the flow rates and steam conditions required by the turbine-generator (TG) supplier.
- Provide backup supply to the auxiliary steam system.
- Supply deaerator pegging steam during startup and shutdown.

10.4 Other Features

U.S. EPR FINAL SAFETY ANALYSIS REPORT

10.4 Other Features of Steam and Power Conversion System

10.4.1 Main Condensers

The main condenser functions as the steam cycle heat sink condensing steam from the main turbine or from the turbine bypass system (TBS).

10.4.1.1 Design Basis

The main condenser performs no safety-related function and therefore has no nuclear safety-related design basis.

The main condenser is designed to meet the following functional criteria:

- Condense steam exhausted from the three low-pressure (LP) turbines. Serve as collection point for steam, demineralized water, equipment drains, extracted water and vented air from other systems.
- Accommodate up to 50 percent of the valves wide open (VWO) main steam flow, which is bypassed directly to the main condenser by the turbine bypass system (TBS). Design features such as baffles, flow distributors, and pressure breakdown devices are included in the main condenser design to preclude component or tube failures due to steam blowdown from the turbine bypass system.
- Receive steam dumped from the TBS without exceeding condenser high backpressure turbine trip (TT) pressure.
- Remove non-condensable gases from the condensing steam through the main condenser evacuation system (MCES), as described in Section 10.4.2.

10.4.1.2 System Description

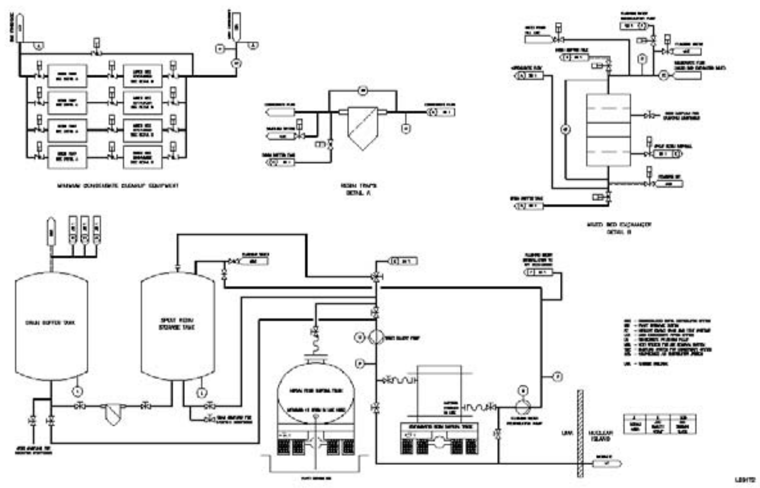
Table 10.4.1-1—Main Condenser Design Data, provides design data. Table 3.2.2-1 provides the quality group and seismic design classification of components and equipment in the main condenser. Section 3.2 describes how the guidance of RG 1.26 is implemented for the U.S. EPR.

The main condenser is a multipressure, three-shell unit. Each shell is located beneath its respective LP turbine. The tubes in each shell are oriented transversely to the turbine generator longitudinal axis.

The three condenser shells are designated as the LP shell, intermediate-pressure (IP) shell and high-pressure (HP) shell. Each shell has two or more tube bundles. Circulating water flows in series through the three single-pass shells, as shown in Figure 10.4.5-1.

U.S. EPR FINAL SAFETY ANALYSIS REPORT

Figure 10.4.6-1—Condensate Polishing System Flow Diagram



Next File

10.4.9 Emergency Feedwater System

10.4.9 Emergency Feedwater System

The emergency feedwater system (EFWS) supplies water to the steam generators (SG) to restore and maintain water level and to remove decay heat following the loss of normal feedwater during design basis transient and accident conditions. This removes heat from the reactor coolant system (RCS), which is first transferred to the secondary side via the SGs, then discharged as steam to the condenser or via the SG main steam relief valves (MSRV).

10.4.9.1 Design Bases

The EFWS provides the following safety-related functions:

- Provide sufficient flow to the SGs to recover and maintain SG water inventory and remove residual heat from the RCS via the SGs and MSRVs to assist in the cooldown and depressurization of the RCS to residual heat removal (RHR) conditions under design basis transient and accident conditions.
- Isolate EFWS flow to the affected SG following a main steam line break (MSLB) to prevent overcooling the RCS with associated positive reactivity.
- Isolate emergency feedwater (EFW) pump flow to the SG with a tube rupture (SGTR) upon SG high water level to prevent SG over-fill and mitigate the potential radiological consequences of a SGTR event.
- Provide sufficient water inventory in the storage pools to support cooldown requirements.

The EFWS has the following design basis requirements and criteria:

- Safety-related portions of the EFWS are designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, and external missiles, and are designed to function following such events (GDC 2).
- Safety-related portions of the EFWS are designed to withstand the effects of the postulated hazards of internal missiles, pipe whipping, and discharging fluids (GDC 4).
- Safety-related portions of the EFWS are not shared among nuclear power units (GDC 5).
- Safety-related portions of the EFWS are capable of bringing the primary plant temperature to the RHR cut-in point following four hours at hot standby from the control room using only safety grade equipment and assuming a single active failure per BTP 5-4 (Reference 1) and GDC 19.
- Safety-related portions of the EFWS have sufficient flow capacity so that the system can remove residual heat over the entire range of reactor operation and

Chapter 10, Steam and Power Conversion System: Emergency Feedwater System

Reliability and Diversity

- **The plant's large primary and secondary inventories provide added time to mitigate the unlikely loss of all EFWS capability.**
  - **Analysis of the postulated complete loss of the EFWS due to common mode failures following a LOOP (without taking mitigating actions) shows that the time to SG dry-out is greater than 1.5 hours and the core remains covered with sub-cooled water for greater than 2 hours**
  - **This extended coping time provides time to initiate compensatory actions which could include**
    - Recovering the safety-related EFWS system and removing core decay heat using the steam generators
    - Removing core decay heat using the plant's safety-related feed and bleed capability
    - Removing core decay heat using the plant's non-safety-related Startup Feedwater system or the plant's non-safety-related Main Feedwater system

Chapter 10, Steam and Power Conversion System: Emergency Feedwater System-Reliability and Diversity

- An overall increase in core damage frequency could result

Risk Measure	PRA Results with 4 MD EFW Pumps	Sensitivity PRA Results with 2 MD & 2TD EFW Pumps
Total CDF [1/yr]	5.3E-07	5.6E-07
LOOP CDF [1/yr]	1.5E-07	1.7E-07
SBO CDF [1/yr]	3.1E-08	3.9E-08

Presentation to the ACRS Subcommittee

AREVA U.S. EPR Design Certification Application Review

Safety Evaluation Report with Open Items

Chapter 10: Steam and Power Conversion Systems

November 19, 2009

Staff Review Team

- **Technical Staff**
  - **Tech Reviewer: Devender Reddy**
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  - **Tech Reviewer: John Honcharik**
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  - **Tech Reviewer: Robert Davis**
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  - **Tech Reviewer: Jeffrey Poehler**
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  - **Project Managers**
  - **Lead PM: Getachew Tesfaye**
  - **Chapter PM: Peter Hearn**

Overview of DCA

United States Nuclear Regulatory Commission
Protecting People and the Environment

SRP Section/Application Section		No. of Questions	Status Number of OI
10.2	Turbine-Generator	7	1
10.2.3	Turbine Rotor Integrity	23	7
10.3	Main Steam Supply System	2	0
10.3.6	Steam and Feedwater System Materials	12	2
10.4.1 10.4.2 10.4.3 10.4.4 10.4.5	Main Condensers, Main Condenser Evacuation System, Turbine Gland Sealing System, Turbine Bypass System, Circulating Water System	5	0
10.4.6	Condensate Polishing System	6	0
10.4.7	Condensate and Feedwater System	3	0
10.4.8	Steam Generator Blowdown System	4	0
10.4.9	Emergency Feedwater System	13	2
Totals		75	12

Description of Open Items

- RAI 329, Question No. 10.02.-7: ITAAC is required to confirm the diversity of the two redundant overspeed protection systems
- RAI 294, Question No. 10.2.03-17: FSAR revision to provide material specifications for procuring the turbine rotors with acceptable fracture toughness
- RAI 294, Question No. 10.02.03-18: FSAR revision to specify the method of calculating the fracture toughness properties of the turbine rotor material
- RAI 294, Question No. 10.02.03-19: FSAR revision to include the HIP in the turbine rotor arrangement sketch, including the weld locations for the HIP and the LP
- RAI 294, Question No. 10.02.03-20: Provide a RAI response that provides information to confirm the integrity of the turbine rotor
- RAI 294, Question No. 10.02.03-21: Revise COL Information Item to ensure the COL applicant submits inspection program and interval information for the staff to review during the COL review.

Description of Open Items

- RAI 294, Question No. 10.02.03-22: Provide ultrasonic inspection of the turbine rotors in the inservice inspection program
- RAI 294, Question No. 10.02.03-23: Confirmation that the analysis of the turbine rotor material properties and the turbine disk integrity applies to the as-built turbine rotor and revise the ITAAC accordingly
- RAI 272, Question No. 10.03.06-11: Provide in a RAI response the weld filler material classification list
- RAI 272, Question No. 10.03.06-12: An explanation for providing a MSSS with a 40 year life in a 60 year life plant
- RAI 305, Question No. 10.04.09-13: Resolution of the conflicting statements in the FSAR dealing with operator actions outside the CR during cooldown
- RAI 238, Question No. 10.04.09-12: Diversity of the Emergency Feedwater pumps power source

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Steam and Power Conversion System

- Turbine Overspeed – D-EHC System
  - In lieu of a mechanical overspeed protection device, AREVA proposed a diverse electrical overspeed device
- Staff Evaluation
  - Tier 1 ITAAC needed to ensure diversity between the two redundant electrical overspeed protection devices
- Open Item
  - Pending review of AREVA response to RAI 329, Question 10.02-7

Materials

- FSAR
  - Rotors fabricated from material with nearest equivalent ASTM A471 specification
  - Tramp elements are controlled, and flaws will be minimized and have improved toughness
- Open Items RAI 294, Questions 10.02.03-17 and -18
  - FSAR should provide:
    - Material specification for procuring turbine rotors
    - Description of procedures to minimize flaws, improve toughness and minimize chemical segregation
    - Method of calculating fracture toughness of turbine rotor material.

Technical Topics of Interest
Section 10.2.3 – Turbine Rotor Integrity (cont.)

Design

- FSAR
  - Rotors are forged and welded
  - Inspection of bores, if applicable
- Open Items RAI 294, Questions 10.02.03-19 and -20
  - FSAR should provide:
    - Turbine rotor design using forgings and welds
    - Location of welds and inspectability, especially for non-bored rotors
    - Operating experience of non-bored rotors, accessibility of inspection and reliability of inspection results
    - Material property degradation of internal region, which are normally removed by boring

Inservice Inspection

- FSAR
  - COL applicants provide inspection interval
  - Visual and surface examination of turbine rotor
  - ITAAC specifies rotor integrity analysis using turbine design and material
- Open Items RAI 294, Questions 10.02.03-21, -22 and -23
  - FSAR should:
    - Provide COL item for providing inspection program in addition to inspection interval
    - Include volumetric (ultrasonic) inservice inspection of the turbine rotors
    - Clarify ITAAC to specify the analysis for the turbine rotor integrity is for the as-built rotor

Technical Topics of Interest
Section 10.3.6 – Steam and Feedwater System Materials

Materials (Background)

- Materials used for Class 2 and 3 components meet ASME Code Section III requirements
- MSSS and main feedwater system fabrication follows applicable Regulatory Guidance (RG 1.37, 1.50, 1.71)
- MSSS and main feedwater system materials meet the applicable fracture toughness requirements

Flow Accelerated Corrosion

- U.S. EPR design incorporates design features to mitigate flow accelerated corrosion (FAC)
  - The following ASME Code and Non-Code systems are designed to mitigate the effects of FAC:
    - MSSS
    - Main feedwater system
    - Condensate system
    - Steam Generator blowdown system
    - Non-safety-related power conversion systems

Flow Accelerated Corrosion – (cont.)

- U.S. EPR design features to prevent FAC include:
  - Materials selection, limits on flow velocity and water chemistry
  - FAC susceptible safety related systems will contain a minimum of 0.10% Cr
  - Susceptible non-safety-related systems may use chrome-molybdenum or stainless steel materials

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Flow Accelerated Corrosion – (cont.)

- COL Information Item 10.3-2 requires COL applicants to develop and implement, prior to initial fuel loading, an FAC monitoring program that conforms to Generic Letter 89-08 “Erosion/Corrosion-Induced Pipe Wall Thinning” and EPRI Technical Report NSAC-202L-R3, “Recommendations for an Effective Flow-Accelerated Corrosion Program.”

Technical Topics of Interest
Section 10.3.6 – Steam and Feedwater System Materials

Materials (Open Items)

- Open Item RAI 272, Question 10.03.06-11 and Open Item RAI 272, Question 10.03.06-12

    - FSAR Table 10.3-11 should specify weld filler materials classifications.

    - FSAR Section 10.3.6.3 states “The minimum design wall thicknesses will be determined in the design phase by the process previously described in order to allow for a minimum lifetime of the affected piping systems of at least 40 years.” The applicant should address why the 40-year design life of the MSSS and main FW system is inconsistent with the design life of the plant which is 60 years.

Technical Topics of Interest
Section 10.4.9 – Emergency Feedwater System

- **Open Item RAI 305, Question 10.4.9-13: Conflicting Information in FSAR Involving Operator Action Outside the Control Room**

  Issue:
  - FSAR sections 5.4.7.3.3 and 10.4.9.3 are inconsistent in whether operator action outside control room is necessary to take the plant to cold shutdown

  Staff Evaluation:
  - SRP 10.4.9 recommends a design conforming to the guidance of BTP 5-4 "Design Requirements for RHR System" in regards to cold shutdown from the MCR
  - Applicant indicated in RAI response that in the event that a EFW pump is unavailable operator action may be required to realign the manual supply header valves to provide access to the inventory for all four storage pools

Technical Topics of Interest
Section 10.4.9 – Emergency Feedwater System

- **Open Item RAI 305, Question 10.4.9-13: Conflicting Information in FSAR Involving Operator Action Outside the Control Room (cont.)**
  - Staff Evaluation (cont.):
    - Staff considers the action of realigning the manual supply header valves operation to be a limited action outside the control room that is acceptable because the action is not required until 6 to 8 hours after EFW start so sufficient time is available for operator action, therefore guidance of BTP 5-4 is met
  - Status
    - To resolve this issue the applicant needs update section 10.4.9.3 to eliminate the inconsistency in the FSAR. A supplemental RAI has been issued

Open Item RAI 238, Question 10.4.9-12: EFW System Diversity

Issue

- The U.S. EPR design uses only motor-driven EFW pumps. SRP 10.4.9 recommends a system design conforming to the guidance of BTP 10-1 as it relates to AFWS pump drive and power supply diversity, and that AFW diversity and performance be reviewed for decay heat removal capability and station blackout capacity

Regulatory Basis

- 10 CFR 50.34 (f) (ii) as it pertains to required evaluation of the AFW system design, capability, and reliability
- GDC 34 and 44 with regards to the requirement of suitable redundancy in components and features, and
- 10CFR50.63 as it pertains to loss of all alternating current power (SBO)

Design Considerations

- The U.S. EPR design incorporates four safety-related, Class 1 EDGs to provide additional redundancy compared to current operating plants, it has two SBO diesels that can power EFWS trains 1 and 4 in the event that both offsite power and onsite EDGS are not available.
- EFWS supply and discharge headers allow any EFW pump to feed different SGs from different storage pools.
- Technical Specification Requirements assure that a minimum of three EDGs will be available or plant is put into 72 hour LCO.
- The EFWS meets the reliability target specified in SRP 10.4.9.

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Design Considerations (cont):

- The SBO diesels provides an alternate AC power source of diverse design, SBODGs differ from the EDGs in model and nominal size, and are physically, electrically, and mechanically separated from the EDGs, They do not share control power, HVAC, or engine cooling, the EDGS are water-cooled, and the SBODGs are air-cooled

Evaluation

- The applicant evaluated the impact that replacing two motor-driven pumps with two turbine-driven pumps would have on PRA, and found that from an integrated PRA perspective, diverse pumps would not be expected to reduce risk significantly.
- For SBO event, SBO diesels are available ten minutes or less into the event and the EFW pumps are assumed to start feeding the SGs thirty minutes into the event

Technical Topics of Interest
Section 10.4.9 – Emergency Feedwater System

Evaluation (Cont)

- The U.S. EPR Steam Generators have sufficient thermal capacity that SG dryout doesn't occur until 1.5 hours after the onset of the SBO event which far exceeds the 30 minute time needed to supply EFW to the SG using SBO diesels
- The staff evaluation confirms that the EFW systems are reliable and capable of providing feedwater to the steam generator under the following conditions: loss of normal feedwater, feedwater line break accident, and station blackout event

Status

- The staff issued a supplemental RAI to have the applicant address concerns about diversity in its licensing basis, and has audited the applicants SG dryout analysis and believe that if the FSAR is adequately updated that we have a path to resolution. The staff recently received the applicant response to its RAI and is currently evaluating it.

ACRONYMS

- AFW – Auxiliary Feedwater
- ASME - American Society of Mechanical Engineers
- ASTM - American Society for Testing and Materials
- BTP - Branch Technical Position
- COL – Combined License
- CR – Control Room
- D-EHC – Digital – Electro-Hydraulic Controls
- EDG – Emergency Diesel Generator
- EFW – Emergency Feedwater
- EPRI - Electric Power Research Institute
- FAC - flow-accelerated corrosion
- FSAR – Final Safety Analysis Report
- FW – Feed Water
- HIP - high/intermediate pressure
- ITAAC - Inspections, tests, analyses, and acceptance criteria
- LP - low pressure
- MSSS - main steam supply system
- PRA - probabilistic risk assessment
- RAI – Request for Additional
- RHR - residual heat removal
- SBO – Station Blackout
- SBODG – Station Blackout Diesel Generator
- SG – Steam Generator

AREVA NP Inc.

Presentation to ACRS U.S. EPR Subcommittee
Design Certification Application
FSAR Tier 2 Chapter 12

Pedro Perez, Supervisory Engineer
Radiological Engineering

Chapter 12, Radiation Protection: Chapter Topics

- 12.1, Ensuring that occupational radiation exposures are as low as reasonably achievable
- 12.2, Radiation sources
- 12.3, Radiation protection design features
- 12.4, Dose assessment
- 12.5, Operational radiation protection program

Chapter 12, Radiation Protection: 12.1, Ensuring that occupational radiation exposures are as low as reasonably achievable

- **Occupational Dose**
  - **Maintenance, in-service inspections, refueling operations**
  - **Radioactive waste handling**
  - **Abnormal plant operations**
  - **Decommissioning activities**
- **The U.S. EPR design reflects on operating experience and implements As Low as Reasonably Achievable (ALARA) principles in the design process**
  - **Physical plant layout that includes compartmentalization and dedicated ventilation**
  - **Material selection reduces activation/corrosion products**
  - **Permanent shielding**
  - **Minimization of contamination following industry lessons learned**
  - **ALARA applied in the design process**

Chapter 12, Radiation Protection: 12.2, Radiation Sources

- **The U.S. EPR radiation sources are derived for normal operations and accident conditions**
- **Source terms were determined for contained and airborne sources of radioactivity**

Chapter 12, Radiation Protection: 12.2, Radiation Sources

- **Normal Operations**
  - The U.S. EPR radiation sources have been calculated following the guidance in the Standard Review Plan Chapter 12
  - Radioactive sources can be contained or airborne
  - The radiation sources are calculated and dose assessments are performed to ensure occupational doses are ALARA
  - A failed fuel fraction of 0.25% is assumed in the reactor coolant source term with iodine and noble gases at the technical specification concentrations
  - Fission products and nitrogen-16 as well as assumed corrosion and activation products constitute the reactor coolant system shielding source term inside the reactor building

Chapter 12, Radiation Protection: 12.2, Radiation Sources

- **Normal Operations (Continued)**
  - **Argon-41 in containment air and tritium in the RCS are also addressed**
  - **The secondary coolant system source term is derived from the transfer of reactor coolant system inventory to the secondary system through assumed steam generator defects**
  - **Radioactive effluent is processed by the gaseous and liquid waste processing systems**
  - **The spent fuel pool is also a radiation source due to assumed fuel defects and corrosion and activation products**
  - **The radiation sources are calculated and dose assessments are performed to ensure occupational doses are ALARA**
  - **The sources are used as well in the design of ventilation systems**

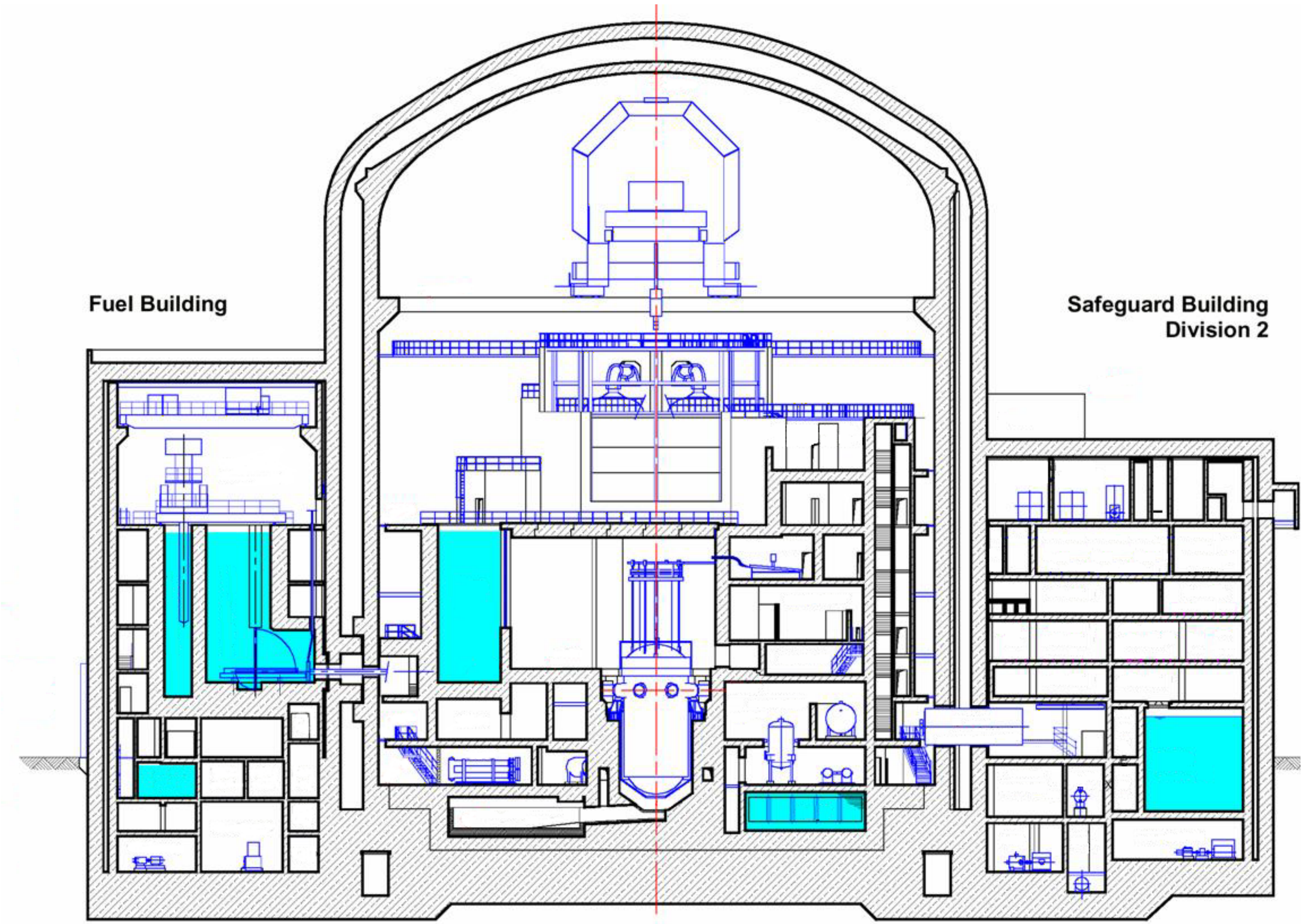
Chapter 12, Radiation Protection: 12.2, Radiation Sources

- **Accident Conditions**
  - **The radiation sources from the radiological bounding accident is used for post-accident shielding and mission dose assessments**
  - **The accident source term is based on the Alternative Source Term design basis loss of coolant accident**
  - **The radiation sources are calculated and dose assessments are performed to ensure mission doses are ALARA**
    - Contained sources in ECCS recirculating liquid
    - Airborne from ESF assume leakage

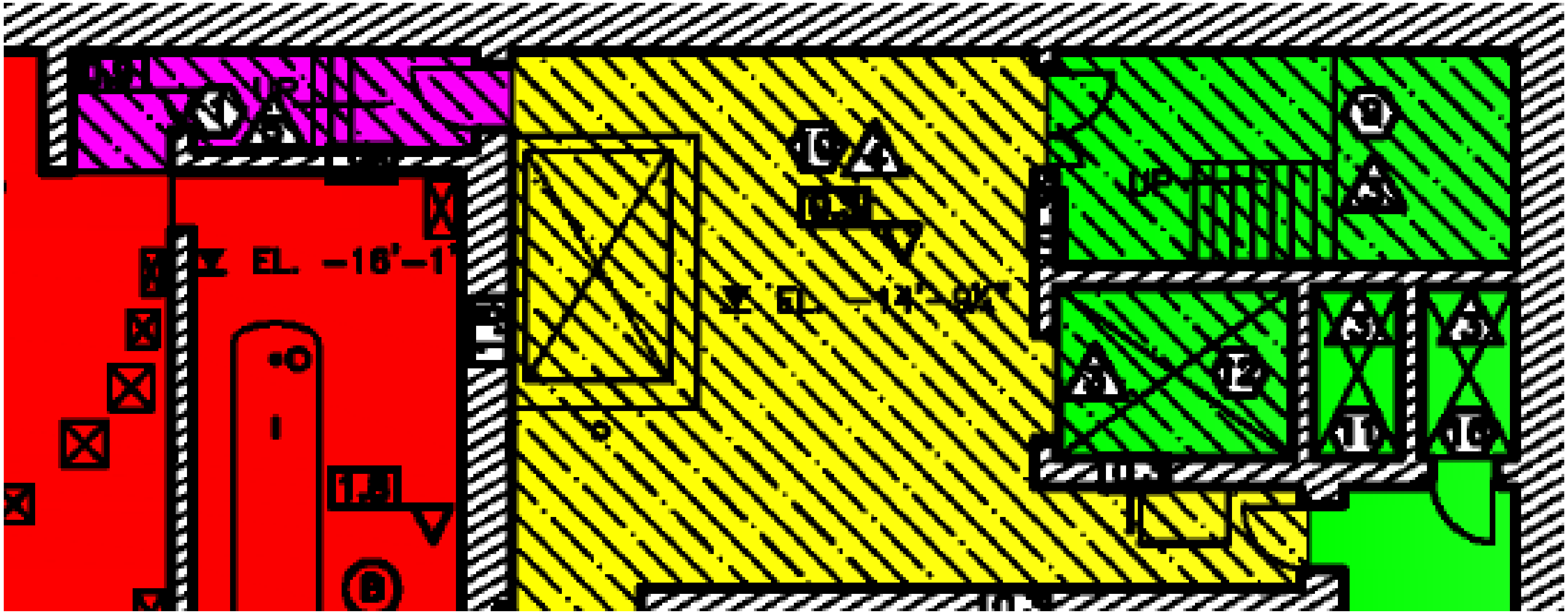
Chapter 12, Radiation Protection: 12.3, Radiation Protection Design Features

- **Occupational and off-site external doses are maintained ALARA by specific design features**
  - **The U.S. EPR Design Considerations – Physical Plant Layout**
    - Compartmentalization creates permanent shields
    - Lower dose areas such as anterooms prior to higher dose rooms
    - Reduction of scaffolding by equipment location or built-in work platforms
    - Segregated systems and compartments
    - System containing radiation sources are placed closest to the reactor building in the lower two floors
    - Two-compartment reactor building
    - Reduction of maintenance exposure by four ECCS trains
    - Ventilation system divided in cells
    - Radiation zones are graduated
    - Radiation zones maintain doses ALARA in general access areas

Chapter 12, Radiation Protection:
12.3, Radiation Protection Design Features



Radiation Zones



Chapter 12, Radiation Protection: 12.3, Radiation Protection Design Features

Reactor Building

- Reactor building consists of a containment building and shield building with an annulus space between the two structures
- The containment building includes a two compartment design with the reactor coolant system and steam generators in the inner compartment
- Outer compartment contains supporting equipment, well shielded from the inner compartment
- Separate ventilation systems process the air in the two compartments

Chapter 12, Radiation Protection:
12.3, Radiation Protection Design Features

- **Safeguard Buildings**
  - **Four separate safeguard buildings house independent divisions**
  - **Each building is divided into a radiological controlled area and an uncontrolled area**
  - **System containing radiation sources are placed closest to the reactor building in the lower two floors**
  - **Two ventilation systems separately serve the controlled and uncontrolled areas**

Chapter 12, Radiation Protection: 12.3, Radiation Protection Design Features

- **Fuel Building**
  - **Divided into cells for ventilation and isolation**
  - **Location of spent fuel pool eliminates a direct pathway to environmental contamination**
- **Auxiliary Building**
  - **Divided into three ventilation cells**
- **Radioactive Waste Processing Building**
  - **Three ventilation cells**
  - **Compartmentalized**

Chapter 12, Radiation Protection:
12.3, Radiation Protection Design Features

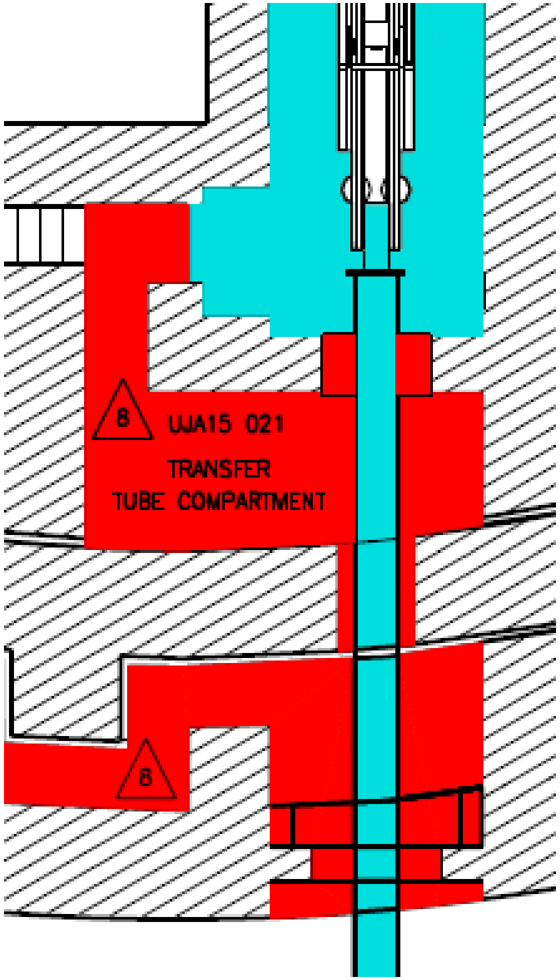
- **Minimization of Contamination (10CFR20.1406)**
  - **Compartmentalization segregates contaminated areas from clean areas**
  - **Potentially contaminated systems are isolated from clean systems by two or more isolating features**
  - **Component leak detection is utilized where there is potential for a direct path to the environment**
  - **Spent fuel pool is away from external walls or floor**
  - **Special features designed to mitigate uncontrolled releases**
    - Steam generator main steam relief train condensation is collected and routed to plant vent and drains controlling potential tritiated liquid releases

Chapter 12, Radiation Protection: 12.4, Dose Assessment

- **Dose rates from components containing radioactive materials were calculated using specific source terms associated with the component; for example:**
  - **Volume Control Tank source term consists of a gaseous top volume and a liquid volume derived from the RCS source term**
  - **Dose calculations for the six room walls are performed with the source geometry**
  - **Fuel Transfer Tube source term consists of the spent fuel radionuclide content**
  - **Dose rate in the annulus space is calculated from the source and geometry**
  - **Dose rates provide the radiation zoning**

Chapter 12, Radiation Protection: 12.4, Dose Assessment

Fuel Transfer Tube (Example)



Chapter 12, Radiation Protection: 12.4, Dose Assessment

Occupational Dose

U.S. EPR Annual Occupational Dose (person-rem)	Average U.S. PWR Annual Occupational Dose (person-rem)*
50	**78**

*Average U.S. PWR annual occupational dose is calculated based upon a three year period average dose from 2005 - 2007 from NUREG-0713 Volume 29

Chapter 12, Radiation Protection: 12.5, Operational Radiation Protection Program

- **The COL applicant that references the U.S. EPR design certification will fully describe the Radiation Protection Program**

Conclusions

- **The occupational dose of 50 person-rem demonstrate that ALARA has been an integral part of the U.S. EPR design process**

Chapter 12, Radiation Protection: Acronymns/Nomenclature

- **ALARA – As Low As Reasonably Achievable**
- **COL – Combined Operating License**
- **ECCS – Emergency Core Cooling System**
- **ESF – Engineered Safety Features**
- **RCS – Reactor Coolant System**

Presentation to the ACRS Subcommittee

AREVA U.S. EPR Design Certification Application Review

Safety Evaluation with Open Items: Chapter 12

RADIATION PROTECTION

November 19, 2009

Staff Review Team

- **Technical Staff**
  - **Sara Bernal**
    Construction Health Physics Branch
- **Project Managers**
  - **Getachew Tesfaye**
  - **Jason Jennings**

Overview of DCA

SRP Section/Application Section		Number of Questions	Number of Open Questions
12.1	Ensuring that Occupational Radiation Exposures are ALARA	0	0
12.2	Radiation Sources	6	2
12.3-12.4	Radiation Protection Design Features	18	7
12.5	Operational Radiation Protection Program	3	1
Totals		26	10

Description of SE Open Items

- **RAI 280 Question 12.02-5:** Incorporate into Section 12.02 of the U.S. EPR FSAR the source term information on the spent fuel, RHR, and Aeroball System provided in response to Questions 12.02-3 and 12.03-12.04-3
- **RAI 280 Question 12.02-6:** Provide clarifying information on the assumptions and equations used to calculate radionuclide airborne concentrations for the reactor and fuel buildings
- **RAI 23 Question 12.03-12.04-1**: Describe how the U.S. EPR will comply with the operational requirements of 10 CFR 20.1406, “Minimization of Contamination.”
- **RAI 228 Question 12.03-12.04-9**: Incorporate into the U.S. EPR FSAR the design feature descriptions, provided in response to several staff RAIs, that demonstrate compliance with 10 CFR 20.1406
- **RAI 228 Question 12.03-12.04-10:** Provide additional detail on the U. S. EPR Demineralized Water Distribution System, including any radioactive systems it interfaces with and how the system design minimizes contamination in accordance with the requirements of 10 CFR 20.1406.

Description of SE Open Items

- **RAI 254 Question 12.03-12.04-15**: Due to Areva's use of the European shielding code RANKERN version 15a, provide sufficient detail such that staff can reproduce the post LOCA dose rate calculations for radiological vital areas located in Safeguards Building 1, using a different shielding code.
- **RAI 280 Question 12.03-12.04-17:** In accordance with GDC 61, describe design features which prevent inadvertent drain down of the refueling cavity via piping or the access hatch located at the bottom of the cavity. Also provide information on the location of a safe laydown area for any fuel located above the reactor vessel flange during inadvertant drain-down.
- **RAI 295 Question 12.03-12.04-18:** Describe the calibration methodology and frequency for the EPR installed area radiation monitors.
- **RAI 296 Question 12.03-12.04-19:** Provide a complete list of Codes used for chapter 12 dose analysis. For those Codes that were modified from the vendor's "as-is" version or that are not publicly available through a U.S vendor, provide information on the QA/QC check Areva applied to the use of these codes.
- **RAI 302 Question 12.05-3:** In accordance with OGC's guidance, the staff requested that COL information item 12.05-1 be modified.

<u>Section 12.1 SER with Open Items:</u>

- **ALARA considerations applied during initial plant design**
- **Equipment design considerations for ALARA**
- **Facility layout considerations to maintain exposures ALARA**
- **COL Information Item**
  - Fully describe elements of the Operational ALARA program for ensuring that occupational radiation exposures are ALARA consistent with 10 CFR Part 20 and the applicable RGs.

Technical Topics of Interest

Section 12.2 – Radiation Sources

Section 12.2 SER with Open Items:

- **Contained source terms for core and major radioactive systems**
  - Spent fuel, Safety Injection System, and Aeroball System (RAI 280 Question 12.02-5)
- **Sources of airborne radioactivity**
  - Clarify assumptions and equations used to calculate radionuclide airborne concentrations for the reactor and fuel buildings (RAI 280 Question 12.02-6)
- **Key SRP Interfaces: 11.1, 12.3-12.4, 12.5**
- **One COL information item**
  - COL applicant will describe any site-specific byproduct, source, or special nuclear material sources over 100 millicuries.

Technical Topics of Interest

Section 12.3-12.4 – Radiation Protection Design Features

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Section 12.3-12.4 SER with Open Items:

- **Facility Design Features**
  - Facility and equipment design features for maintaining exposures ALARA
    - Spent fuel transfer tube
    - Applicant should describe design features which prevent inadvertent drain down of the refueling cavity (RAI 280 Question 12.03-12.04-17)
  - Plant shielding design
    - Confirmatory analysis
  - Ventilation system design to minimize personnel exposures

Technical Topics of Interest

Section 12.3-12.4 – Radiation Protection Design Features

Section 12.3-12.4 SER with Open Items:

- **Facility Design Features (continued)**
  - Area radiation & airborne radioactivity monitors
    - Describe the calibration methodology and frequency for the EPR installed area radiation monitors (RAI 295 Question 12.03-12.04-18).
  - Post-accident access
    - Contribution of airborne radioactivity to mission dose
  - Key SRP interfaces: Chapters 7, 9, 11, and 12

Technical Topics of Interest

Section 12.3-12.4 – Radiation Protection Design Features

Section 12.3-12.4 SER with Open Items:

- **Dose Assessment**
  - Dose-reducing measures and design modifications
  - Projected annual exposures
- **10 CFR 20.1406, "Minimization of Contamination"**
  - Provide additional detail on the U. S. EPR Demineralized Water Distribution System (RAI 228 Question 12.03-12.04-10).
  - Operational requirements of 10 CFR 20.1406 (RAI 23 Question 12.03-12.04-1)

Section 12.3-12.4 SER with Open Items:

- **Three COL Information Items**
  - Provide site-specific information on sampling, recording and reporting airborne releases of radioactivity.
  - Estimated annual doses to construction workers as a result of radiation from onsite radiation sources from the existing operating plant(s).
  - Describe the use of portable instruments to determine the airborne iodine concentration during an accident, in accordance with requirements of 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737.

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Section 12.5 SER with Open Items:

- **Operational Radiation Protection Program**
  - COL applicant will fully describe the Operational Radiation Protection Program, including organization; equipment, instrumentation and facilities; and procedures – COL information item 12.05-1

ACRONYMS

- SE – safety evaluation
- OGC – office of the general counsel
- RAI – request for additional information
- COL – combined license
- RG – regulatory guide
- GDC – general design criteria
- SRP – standard review plan
- LOCA – loss of coolant accident
- ALARA – as low as is reasonably achievable