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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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APR1400 SUBCOMMITTEE

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THURSDAY

SEPTEMBER 22, 2016

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ROCKVILLE, MARYLAND

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

COMMITTEE MEMBERS:

RONALD G. BALLINGER, Chairman

CHARLES H. BROWN, JR., Member

JOSE A. MARCH-LEUBA, Member

DANA A. POWERS, Member

JOY REMPE, Member

PETER C. RICCARDELLA, Member

JOHN W. STETKAR, Chairman

MATTHEW W. SUNSERI, Member

DESIGNATED FEDERAL OFFICIAL:

CHRISTOPHER L. BROWN

ALSO PRESENT:

TONY AHN, KHNP

MAITRI BANNERJEE, NRR*

JOHN BUDZYNSKI, NRO

BOB CALDWELL, NRO

SEOG NAM CHOI, KHNP

WOOCHONG CHON, KEPCO NF

JEFF CIOCCO, NRO

TIM DRZEWIECKI, NRO

SONG DUK BIN, KEPCO E&C

GORDON HALL, Westinghouse

JOHN HONCHARICK, NRO

JASON HUANG, NRO

GYUN DO JEONG, KHNP

REBECCA KARAS, NRO

JONG SOO KIM, KHNP

JUNGHO KIM, KHNP

MINSEOK KIM, KHNP

YUN HO KIM, KHNP

LIM KWANGIL, KEPCO E&C

SUNGUK KWON, KHNP

HIEU LE, NRO

CHANG-YANG LI, NRO

GREG MAKAR, NRO

RYAN NOLAN, NRO

JIYONG OH, KHNP

ERIC REICHELT, NRO

JAMES ROSS, KHNP/AECOM

ROB SISK, Westinghouse

TAE SUN RO, KEPCO E&C

ANGELO STUBBS, NRO

JEONG KWAN SUCH, KHNP

YOUNG SIL SUL, KEPCO E&C

JESSICA UMANA, NRO

DAN WIDREVITZ, NRO

SEOK JEONG YUNE, KEPCO E&C

* Present via telephone

A G E N D A

P R O C E E D I N G S

8:35 a.m.

CHAIRMAN BALLINGER: This meeting will come to order. This is a meeting of the -- a renewed meeting of the APR1400 Committee of Advisory Committee of Reactor Safeguards. I'm Ron Ballinger, Chairman of the Committee. Members in attendance are Joy Rempe, Charlie Brown, Jose March-Leuba, John Stetkar, Matt Sunseri, Dana Powers and Peter Riccardella.

The purpose of today's meeting is an extension of yesterday's meeting to receive briefings from KHNP regarding -- and the staff regarding the design certification for the APR1400. The rules for today's meeting again were published in the *Federal Register* on September 7, 2016. The meeting was announced as an open and closed meeting, so if we have questions that turn out to be proprietary, you need to let us know so we can close the meeting.

A transcript is being kept and made available in the *Federal Register*, as stated in the *Federal Register* notice. We would request that people shut off or disable chicken sounds and stuff like that various cell phones and other items that we've heard already this morning.

Participants should first identify themselves when they have comments, and speak with sufficient clarity and volume so that they can be readily heard. We have a bridge line open I think again, and before we get started, I want to make some clarifications of yesterday.

This is a subcommittee meeting, which means that questions from members are questions from the subcommittee members themselves. They're not questions from the ACRS. The ACRS communicates through its letters only.

So while we have kept a list of questions and we've gotten asked questions where the applicant or the staff may say we'll get back to you because we need to get an answer, that is not an RAI all right, definitely.

Not an RAI, and so we'll adjudicate those questions amongst the staff and things like that. So that's to be very, very clear about that and I think that's clear under Member Stetkar wants to say something in addition.

MEMBER STETKAR: No, that's pretty clear.

CHAIRMAN BALLINGER: So with that, does

-- you want to say something Bill or Jeff?

MR. CIOCCO: Jeff Ciocco. I'm the lead project manager for the APR1400 design certification project. Thank you for having us back this morning. We look forward to giving you our staff presentation on Chapter 5. We have everybody assembled up here at the table. Over here, we had good representation by our technical staff in the audience, as well as our technical branch chiefs, and we have a division director with us as well today from the Division of Engineering, Bob Caldwell.

So we look forward to giving you our presentation, any questions that you have, thank you.

CHAIRMAN BALLINGER: And so who's up? Jessica.

MS. UMANA: Okay. Good morning. I'm Jessica Umana. I'm the Chapter 5 Project Manager for APR1400. Today, you will hear the staff's presentation to ACRS on reactor, I'm sorry, reactor coolant system and related systems. Specifically, the staff will cover technical topics and open items that came up fast in our review, which was captured in the safety evaluation which you received some time prior to this meeting.

I'd like to take this opportunity today to just present the presenters. We have John Budzynski, Dan Widrevitz, John Honcharik and Greg Makar. Let me move on through the slides. While you see the presenters up here, I do want to take a moment to appreciate this item. The next is the entire staff that worked on Chapter 5.

On this slide, we have it over on this one and I believe the next two slides we an overview of the SRP sections that were reviewed and the number of questions that were generated from each section with open items. There is one correction I'd like to make, and that's on this slide.

This one shows that we have one open item in 5.4, but that open item is actually in Section 5.4.1.2. The confusion was on my part based on the numbering scheme that was produced when the RAI went out. But I do want to make that clarification. It's not in 5.4. It's 5.4.1.2.

I believe here we have a summary. Overall, 77 questions were generated in Chapter 5, and we have 13 open items. I think maybe one or two may have been brought to resolution since this presentation was prepared by the -- I'll let the staff cover that.

So as you can see here, go back a few slides. A number of questions were generated in 5.4.1.1, which is the Reactor Coolant Pump Flywheel, and then we had a few questions come from 5.2.3. So now we'll turn it over to John Budzynski. He's going to be presenting in 5.2.2, which is Overpressure Protection, 5.4.1.2, Reactor Coolant Pumps, 5.4.7, Shutdown Cooling, 5.4.12, Reactor Coolant System High-Point Vents.

MR. BUDZYNSKI: Okay. The first section we're going to hear is Section 5.2.2, Overpressure Protection System. The main primary reason for the overprotection is to protect the reactor cooling system, connecting systems and secondary sides of steam generators.

The reactor cooling of the pressure boundary is consistent with four POS RVs, two shutdown cooling relief valves and 20 main steam safety valves, five per steam line and the shutdown cooling suction line relief valve provides sufficient pressure, relief capacity to indicate the most, the main low pressure overprotection event.

Let's go on to the next slide. In the there open items, two have been since closed. The only remaining open item is the second one, 5.2.2.7 and we requested additional information on how they determine the methodology in what computer codes they used, and then put parameters in the assumptions being made in the analysis for a limiting LTOP event.

MEMBER STETKAR: John, I had a question about that, juts for my edification. When I read through that section of SER, I know you're concerned about energy input from the secondary to the primary system. Is that what --

MR. BUDZYNSKI: Yes.

MEMBER STETKAR: It's noted. It says "The applicant did provide information on the secondary to primary heat transfer using the secondary temperature 230 degrees Fahrenheit, 110 degrees C greater than the RCS, which the staff concludes is substantially more conservative than the 100 degrees Fahrenheit, 37 degrees C difference allowed by the technical specifications, and is therefore acceptably conservative."

How come we have a situation where the secondary temperature is at 230 degrees Fahrenheit higher than the primary temperature under conditions when LTOP is required, and I don't -- because yesterday I couldn't ascertain whether or not the actual enable temperature was proprietary. So we all know what it is but --

MR. BUDZYNSKI: I would have to say that the gentleman that wrote this RAI is not present, and he is the gentleman that's familiar with this RAI. I would have to go back and discuss it with him --

MEMBER STETKAR: Because that -- see, that's a really big delta T --

MR. BUDZYNSKI: Yes, it is.

MEMBER STETKAR: And I'm not a thermal hydraulics guy, but I sort of know where water boils under what different pressures and how heat transfer normally works.

So it's really hard for me -- when you say it's acceptably conservative, if it's -- if it's absurd, that isn't what we ought to be doing either. So if you could help me with that somehow, you know, just put it on -- it's on the transcript and as Ron said it's just a question.

MR. BUDZYNSKI: Okay.

MS. KARAS: This is Becky Karas.

MEMBER STETKAR: It's only -- I'd hate to see applicants doing analyses with just artificial conditions, and then somebody saying well that's acceptably conservative.

MS. KARAS: This is Becky Karas, chief of Reactor Systems. I can try and get the reviewer down here.

MEMBER STETKAR: Oh, it would help if it's a quick answer, you know. If it isn't, then we're going to be meeting on these things.

MS. KARAS: Okay, okay, all right.

MEMBER STETKAR: It isn't -- it doesn't necessarily have to be answered, you know, immediately if the cognizant person isn't here.

MS. KARAS: Yeah. I can try and get him down here during this meeting, okay.

MEMBER STETKAR: Thank you.

MS. KARAS: Okay.

MR. BUDZYNSKI: The next section 5.4.1.2, Reactor Coolant Pumps. The pump design is -- the reactor coolant pump is a vertical shaft, single stage centrifugal pump. Loss of cooling water event is the most serious one. CCW flows to the pump motor bearing old cores, motor air coolers, high pressure seal water coolers pump and the CVCS continues to provide sealed water injection flow to the RCP pump by seals.

Pump and motor bearings designed to withstand the loss of CCW for 30 minutes. CCW, that's Division 1, supports charging pump new flow heat exchanger and CVCS and all coolers under RCPS. The seal design is improved to have low seal leakage rates during a station blackout event.

Two open items that have since been closed and resolved so we won't discuss it here.

MEMBER STETKAR: I had, John, a question on the overspeed for the flywheel, and I hit you with it because I beat up the applicant enough yesterday. When I read their section of the DCD, they're very, very, very careful in the DCD to say that the highest anticipated overspeed is predicted for a loss of coolant accident, with the largest break size remaining after the application of leak before break.

That break size is apparently four inches, okay. There's a lot of much bigger piping connections to the reactor coolant system than four inches. How do I treat -- I mean does the staff accept leak before break arguments in terms of LOCA sizes for this type of analysis? What happens if I get like a 12 inch break? Does the flywheel come apart? That's my concern.

(Off mic comments.)

MR. HONCHARIK: I guess this is John Honcharik. So I guess the question is whether or not that four inch line is the limiting line?

MEMBER STETKAR: They claim it's the limiting line after they apply all of their leak before break stuff. In other words, whatever -- I haven't read Chapters 3.6.3, nor am I a specialist in leak before break analyses because that's always mystified me. But apparently after you do your leak before analyses, they conclude that the largest break equivalent size that they can get after all of that is four inches, and they did their overspeed analysis based on a four inch break is my understanding.

It said based on that four inch break, I won't exceed my design overspeed of the pump, and my question is well, you know, the loops themselves are really big and I don't want to get into the double ambiguity and sheer of the loop, and there's pressurize the surge line it's 12 inches.

The direct vessel injection lines are like eight or eight and a half inches. Couldn't find the charging and letdown lines sizes, but they're probably, you know, in the four inch or so range. What happens if they get a bigger break, you know? Essentially you've accepted their overspeed analysis as being acceptable.

What I don't know is if they have a bigger break, can they get a larger overspeed and have the flywheel come apart?

MR. HONCHARIK: Yeah, because I looked at the flywheel and I know, I mean I know they -- I think they -- I'm sure they did leak before break as you stated, and basically there, instead of having a break you'll have a leak. So more than likely those, what they're saying then is that those larger pipe sizes more than likely will have a leak, not a break.

So some of the limiting ones would be the other ones, and I think the largest that they specified was four inches. But I'd have to consult with someone for the leak before break.

MEMBER STETKAR: Yeah, and indeed that's what they say they did. But what I'm curious about -- and I don't know how that factors into design basis LOCA analyses. That's a different topic. I'm just saying for the purposes of evaluating the integrity of the flywheel or an overspeed is does the staff typically accept those types of leak before break arguments.

I'm not sure that I've seen that before in other applications, but I must admit that I didn't go back and try to do a thorough search of Chapter 5 to see how people did those overspeed analyses.

MR. MITCHELL: Yeah, and this -- if I could interject. This is Matthew Mitchell, Chief of the Materials and Chemical Engineering Branch in NRO. I believe the characterization with respect to the span of systems which on occasion be applied a leak before Break 2 is accurate and that they did eliminate from a leak before break analysis perspective lines down to -- essentially into the forward trench.

I think you will find that in 363 when we get to covering that. From a philosophical standpoint, in terms of the application of leak before break technology, I think the staff should probably take this back and discuss this.

MEMBER STETKAR: I was more -- the reason I asked you rather than KHNP is that they have used the leak before break argument in a few places throughout the DCD, and to me this is, you know, it's partly technical, it's partly philosophical. Because you've -- this is first place where I could hang my hat on something that the applicant has explicitly said they've accounted for, it's why I'm kind of raising it now in terms of what's the staff's philosophy in accounting for those arguments.

MR. MITCHELL: And I completely understand the question. I'm sure, as you will recollect, leak before break has inherently been tied to general design criterion 4, and the elimination of the dynamic effects of pipe rupture.

MEMBER STETKAR: Exactly.

MR. MITCHELL: And that has since the initiation of the leak before break concept been a rather well-defined regulatory box in terms of the type of effects, jet impingement, pipe whip, that could be essentially excluded from the design basis based upon the demonstration of leak before break.

MEMBER STETKAR: Yes.

MR. MITCHELL: Having done leak before break analyses myself for many, many years, back when I was in the other office and then my branch deals with LBB now, I will say that I'm not familiar with a prior application that I can recall where it would have been associated with flywheel overspeed.

I'd need to go back and research that a bit more to see whether or not that was ever established by precedent. But that would be within the scope of dynamic effects of pipe rupture.

MEMBER STETKAR: Yeah. I didn't, as I said, I didn't try to go back to the other DCDs that I have and track it, only because I ran out of time. But I didn't recall hearing it and I've sat in on several of the design certifications over the last seven-eight years or so, and I didn't recall hearing that type of argument before.

MR. MITCHELL: And I think I would agree that we probably want to go back and discuss it amongst ourselves a little bit more because I, likewise I'm not sure whether I've run across that.

MEMBER STETKAR: Yeah. Okay, thank you.

MS. KARAS: Dr. Stetkar, at whatever point you want to return to your prior question, we have the reviewer here for that.

MEMBER STETKAR: I'm pretty thrilled right now before I forget what it was.

MS. KARAS: If you could possibly restate your question, it would be helpful.

MEMBER STETKAR: Oh gee. Well, now we're really pressing. Let me go back to where the heck it was.

CHAIRMAN BALLINGER: Oh, the temperature on the --

MEMBER STETKAR: No. In all seriousness, the concern as I understand it for the energy and for the low temperature overpressure protection analysis was concerning with heat transfer from the secondary side to the primary side, energy transfer from the secondary side -- there you are -- to the primary side.

In the SER, there's a statement that said you're still tracking that as an open item. We know that. So the applicant did provide information on the secondary to primary heat transfer using the secondary temperature of 230 degrees Fahrenheit, 110 degrees C greater than the RCS, which the staff concludes is substantially more conservative than the 100 degrees Fahrenheit, 37 degrees C difference allowed by the technical specifications, and is therefore acceptable and conservative.

My question was for your benefit, because you may not have been here, I don't know whether the LTOP enable and disable temperatures are proprietary information, so I didn't cite those temperatures. We all know what they are. But given the enable temperature, is it feasible at all to have a secondary temperature 230 degrees higher than that temperature? In other words, how does thermal dynamics work in this situation?

MR. DRZEWIECKI: So there's another event, I'm sorry. Is this in?

MEMBER STETKAR: Well, I think so, yeah.

MR. DRZEWIECKI: It's on, okay. I'm Tim Drzewiecki. I'm from the Systems Branch. We also looked at this in an event in Chapter 15 as well, in which we were looking at some of the overpressure events there. As far as that specific answer to your question, I don't it my fingertips. I can follow up on that.

It was addressed in an RAI because actually the event that I'm thinking of is a startup. It's a startup of an RCP in a random loop. So we asked an RAI on that, trying to get the basis for where those values are from. I can follow up with that RAI value there, but it's addressed as well in Chapter 15. I forget the event number.

However, as far as the tracking of this RAI and why it's an open item, it is that we were not clear on how the calculation was actually done, some of the other inputs and things like that.

That's why it's being tracked as an open item.

MEMBER STETKAR: My question when I stumbled across the actual temperature was, you know, I understand having confidence that you, that the staff must understand how the applicant did their analyses. That's clear. There's no question about that, and it's also clear that the analysis should be appropriately conservative.

But it shouldn't necessarily be unrealistic in the sense of you can't achieve these conditions. In other words, just saying that, you know, I could pick a 1,000 degree temperature difference and that would be even more conservative, but we all know that's physically impossible.

So I was just questioning, you know, what basis. Are there conditions where you could get that type of delta T in that direction at the time when you need LTOP. It's clear there are other conditions where maybe you could get that type of delta T during a cooldown, while the secondary and the primary system is still intact, but now when you're below the -- as cold as you are when you get the LTOP enable or even the disable temperature.

MR. DRZEWIECKI: Yes, I understand that.

MEMBER STETKAR: Because you're talking

about a few hundred degrees on the secondary side.

MR. DRZEWIECKI: Yeah, yeah. So as far as the details, I'm sorry, I don't have it at my fingertips.

MEMBER STETKAR: No, that's fine.

MR. DRZEWIECKI: But it was definitely addressed in the section that's within 15 on the startup --

MEMBER STETKAR: On the startup of the pump.

MR. DRZEWIECKI: Yeah.

MEMBER STETKAR: Okay, because that again, that might have been. I don't want to speculate. I didn't go over in Chapter 15 and try to draw this. I just found it Chapter 5.

MR. DRZEWIECKI: Yeah, okay.

MEMBER STETKAR: Okay.

MR. DRZEWIECKI: Thank you.

MEMBER STETKAR: Thank you.

MEMBER REMPE: Excuse me while you're still there. I'm a little confused on what it is you're still investigating, because what's in the draft DIC says well, the inputs are conservative and I thought I heard today that it's not the inputs anymore; it's just the methodology you're concerned about at this point or what is it?

MR. DRZEWIECKI: So as far as the language that the inputs, we found them to be -- conservative and acceptable. I really can't speak to that. I don't know if Gordon wrote that. However, the one thing that I will say is as far as the methods and the actual calculation that was done, that's what we have questions on.

MEMBER REMPE: So this is that it was the code. It starts with an O, OBERP, O-B-E-R-P is the name of the code that was used, right?

MR. DRZEWIECKI: It could be.

MEMBER REMPE: Okay, but that -- okay. So I was just looking at the RAI and the response back, and I was just kind of curious on what was the problem. But I guess we'll hear about it later.

MR. DRZEWIECKI: Okay. Just their response to that was the 5.2.2.7 follow-up, because that would be the one that's still on the evaluation.

MEMBER REMPE: Right.

MR. DRZEWIECKI: Okay.

MEMBER REMPE: But there's two parts to it, and I assume that the hand calculations aren't causing the problems when you're talking about the methods, because that usually isn't a problem with a hand calculation. But the energy, the second one talks about the method of analysis. But anyway, I just am curious about it. But I guess we'll get an update and we'll hear about it later is the answer to the question at this point.

MR. DRZEWIECKI: Yes.

MEMBER REMPE: Thank you.

MR. BUDZYNSKI: Okay. We're on 5.4.7, Shutdown Cooling. Configuration similar to most current PWR designs. Two independent trains. Shutdown cooling pumps are interchangeable with the containment spray pumps. So this is used during shutdown, refueling, startup and it's used in the range of approximately below 450 psig and 350 degrees Fahrenheit.

When we did the evaluation of gas accumulation, we used Generic Letter 2008, RIS 2013-09 and we evaluated it also against NEI 09-10. We have one open item. We accepted the ITAAC that they wrote, but the only problem we have is that the NEI 09-10 was not referenced in the DCD, and we wanted it to be referenced in the DCD because it -- more or less it's the most current gas management procedure we have, guidance we have.

It takes into consideration void testing and it takes into consideration of a gas accumulation high point sections of the piping configuration, and it's all put together one piece and we wanted that to be referenced. As far as the ITAAC, it was acceptable and we can close that out within -- once we get together with the applicant.

MEMBER REMPE: So I have a question about the section with an RAI that was closed out. It's RAI 381-8100 about natural circulation.

MR. BUDZYNSKI: Yes.

MEMBER REMPE: And the response back from the staff. The reason I'm interested in this is just for future evaluations when we look at accident analysis and natural circulation in a CE plant, and I guess they're basing things on some Palo Verde data and in some startup testing and I just was curious. Are you the person who evaluated that?

MR. BUDZYNSKI: Yes, yes.

MEMBER REMPE: Okay. So could you elaborate a little bit more about why you find that acceptable?

MR. BUDZYNSKI: Okay. What we initially looked at, there was a decay heat ratio of less than .87 I believe it was, and when we have some hazard they come up with that number. We could find it anywhere. If they said it was less than that, then it's acceptable. There would be no problem in natural flow conditions, okay.

They responded and when they responded they showed that it would demonstrate that at full power, the amount of heat being produced for -- I'm trying to -- I don't have the question in front of me, so I'm trying to think of the response.

MEMBER REMPE: It's okay. A good approximation.

(Simultaneous speaking.)

MR. BUDZYNSKI: --heat produced at full power versus the amount of heat produced at the natural flow conditions, recirculation conditions would be less. So therefore you're not exceeding any type of thermal limits at that condition. Basically, that's what they're saying.

And to me, that made sense, but I don't have my notes in front of me to go into detail on that, yeah.

MEMBER REMPE: Well, I pulled the NUREG from the email that did the calculations for this or the testing and the results, and so I guess we'll be discussing this a little bit more as we go further into the DCD in other sections.

MR. BUDZYNSKI: Yeah. I can get better information that I have on my desk on that.

MEMBER REMPE: Okay, thank you.

MR. BUDZYNSKI: Okay. RCS high point events. There's no open items here. This is just a description of the technical topics that were covered during evaluation. Do you want me to go over those, or they're just listed with it.

MEMBER STETKAR: I have a question, but it doesn't pertain to any of the bullets here but it does pertain to the vents so --

MR. BUDZYNSKI: Okay, go ahead then.

MEMBER STETKAR: In the SES, this is -- I need to pretty much understand what you thought about it. In the SER, you start off saying the staff reviewed Section 5.4.12 in accordance with the standard review plan. To ensure the adequacy of the RCGVS, to remove from high points in the RCS non-condensible gases that could hinder natural circulation and core cooling after a design basis event.

Later on in that discussion, it says "To address potential gas accumulation in the steam generator tubes, the applicant described a procedure to operate the reactor coolant pump for short periods of time to transfer non-condensible gases from the tubes to the other RCS high points for venting, therefore in accordance with 10 CFR 50.34(f)(2)(vi) and 10 CFR 50.46(a). Vents are provided for the RCS high points for the APR1400.

I went back and I read the DCD pretty carefully, because I stumbled over that discussion in the DCD when I first read it, and I later concluded that their discussion of using the reactor coolant pumps was completely unrelated to a natural circulation event, which to me makes a lot of sense because, you know, I kind of have natural circulations when I don't have the reactor coolant pumps.

MR. BUDZYNSKI: Right.

MEMBER STETKAR: So I can't bump the reactor coolant pumps to take non-condensibles out of the steam generators during a natural circulation event. You may want to go back and look carefully at the way the SER is written, because I came away from the SER with the conclusion that natural circulation cooling is fine, and the vents are fine because they can bump the reactor coolant pumps to take gas out of the U tubes during a natural circulation event.

MR. BUDZYNSKI: Okay. I'll take a look at that.

MEMBER STETKAR: And if that -- if that was your intent, I'm very curious about how one can actually do that. As I've said, when I read the DCD, the way the DCD is organized they're really careful to say that they can open the vents to remove non-condensibles from the pressurizer and in particular from the reactor vessel head I think is the only time that they used the words natural circulation.

They talk about bumping the reactor coolant pumps as a way of degassing the system in another part of the discussion, which is more oriented toward normal operation, normal venting of the system, for example, for cooldown or heat up or things like that. So I'm hoping it's just my reading the SER wrong, and that you're not actually taking credit for that process, to facilitate natural circulation cooling.

MR. BUDZYNSKI: Okay. I'll make that an item to go back and talk to the engineer that did the review.

MEMBER STETKAR: It could be a wording situation in the SER, or if it's actually a technical conclusion, I'd like to understand how they came to that.

MR. BUDZYNSKI: Okay.

MS. UMANA: Okay. We have Darren Widrevitz up to present reactor coolant pressure boundary materials.

MR. WIDREVITZ: So in terms of this slide, I take the open item was an editorial concern that we were looking at a little bit later in the game. I think there really is no technical significance to it. So unless you had questions about it, I'd like to entertain questions about the section in general. Does that make sense?

There was -- the text of the DCD implied a particular -- some welding parameters were sufficient to demonstrate non-sensitization, when in fact it was a coupling with the ACM and A262 that's ANE (phonetic), and using these parameters that they felt was sufficient to produce non-sensitized product.

CHAIRMAN BALLINGER: I think we went through that to some extent with the applicant yesterday on this A262 and where it's used and I think we concluded, at least I did, that Practice A262A and E are -- would never be used on an actual weld.

They might be used as part of weld qualification as a Member Riccardella mentioned, but not as part of any kind of QA on an actual weld on a pipe, since Practice E requires a U bend sample to be taken out of -- to be taken out in Practice A would require basically inducing IGA in the weld itself.

MS. UMANA: Okay, all right. We'll move on. We have Section 5.2.5 on Reactor Coolant Pressure Boundary Leakage Detection, and I'll be covering this slide. Developing requirements of this area are found in GDC 30 and GDC 20, with the acceptance criteria outlined in Reg Guide 1.45 for GDC 30 and Reg Guide 1.29 and 1.45 for GDC 2 (sic).

The staff conducted its review against SRP Section 5.2.5. I'm sorry, the following review -- the review areas are listed here for you. I'll spare you reading all of those. This one was pretty straightforward. There were three RAIs, most requesting clarification and resolution of some inconsistency, and all were statused as confirmatory items. I'm hoping to see those changes in the next revision to the DCD.

Now there was one COL information item and the staff found it acceptable, because it's consistent with the Reg Guide which is referenced in the COL information item. Do you have -- if you have any questions, I'll defer to the technical reviewer in the audience.

CHAIRMAN BALLINGER: Maybe this is a time to ask a question which I wanted to ask yesterday, and it's related to this but unrelated to this. I searched for, because I'm sensitized, if you will, to steam generator tube leaks. So I'm always after looking to see what is the allowable, unidentified and identified leakage.

And so that number is bandied about in the DCD and in one place it says .5 gallons per minute is unidentified leakage is allowed, and then ten gallons per minute identified leakage. Is that correct?

MS. UMANA: Chang Li?

CHAIRMAN BALLINGER: Because I've seen other numbers in the document and I'm thinking --

MS. UMANA: I have the technical reviewer here. He'll be able to answer that.

CHAIRMAN BALLINGER: And then 150

gallons per minute through one steam generator.

MEMBER STETKAR: That's a typo.

CHAIRMAN BALLINGER: That's a typo? That has to be a typo, okay. I was going to say that.

MR. LI: I am the reviewer for this sections. What you said is correct for unidentified leakage, the limit specified in the tech spec is .5 gallons per minute. For identified leakage is 10 gallons per minute, and there's an interface between steam generator. In steam generator leakage, I think you quote 150 gallons per minute?

CHAIRMAN BALLINGER: Yes.

MR. BUDZYNSKI: That's per day in the tech specs.

CHAIRMAN BALLINGER: Per day.

MR. BUDZYNSKI: There's a typo in the SER.

MR. LI: The typo, yeah.

CHAIRMAN BALLINGER: Okay, thank you.

MS. UMANA: If there are more questions, we're going to move onto -- I think this is a topic that you enjoy very much, the flywheel integrity and I'm going to hand it over to John, and then he can go through his --

MEMBER STETKAR: I apologize for that. I just somehow associate the flywheel with the reactor coolant pump. I didn't look far enough in your presentation, so sorry.

MR. HONCHARIK: Hi, good morning. I reviewed the reactor coolant pump flywheel integrity, and basically the KHNP provided some areas in accordance with the Reg Guide, and that included material selection and fabrication technique for the flywheel, and any pre-service inspection that will be done and also in-service inspections, and also the overspeed testing per the SRP.

Also they provided the flywheel analysis for the Reg Guide, and that basically looked for the critical speed for the ductile and non-ductile fracture mechanisms, and also it included a fatigue crack growth rates for the flywheel.

Next slide. As everyone stated, there were six open items for this area. The first one was basically they didn't specify what method they're going to determine the fracture toughness of the material in question. So basically we asked that. I think they -- per yesterday, they responded about I think a week ago. So we're still evaluating that, but I think they mentioned something that they're going to use the direct method basically to do testing for it.

The next item was the operating experience. We have some operating experience for the material that they were using. Mostly it was clarifications or what actual operating experience was applicable to this material and situation. So I think they also provided that.

Next was the -- right now the applicant provided an analysis using one-third ultimate strength. However, the SRP specified that they used a one-third yield strength. So basically we asked a question okay, if you're going to use one-third of the ultimate, provide justification why that's acceptable. If not, use the one-third yield per the SRP.

I think they responded, I think they're probably going to use the one-third yield. They're going to --

MEMBER STETKAR: There wasn't a big difference if I recall, was there?

MR. HONCHARIK: Huh?

MEMBER STETKAR: There wasn't a very big difference between those two values, was there?

MR. HONCHARIK: Probably not, yeah.

MEMBER STETKAR: Yeah.

MR. HONCHARIK: So they're going to do that and I think November they're supposed to provide that revised analysis. The other thing that they did was this flywheel is pretty much similar to all the other ones, except this one has hub, some place for a key. They have that hub that attaches it to the rotor.

So basically we're asking that they also do an analysis of that hub, you know, because if that fails, it could potentially release the flywheel. So I think they responded and they're going to do analysis and I think basically they're basing a lot of it on all that's going to be compression. So that will be coming in November along with the other analysis.

And also any inspections that you're going to be doing on that hub should be the same as the flywheel. So they're also responding to that.

The next was the -- this was just a minor one. Basically, they needed to specify what the acceptance criteria for the NDE methods, so that the acceptance criteria for flaw size, it will be bounded by the analysis. So we're just looking to make sure that that was consistent and they respond to that. I think they're going to put that in the DCD.

Next slide, and the last one was basically to put in the DCD, because they didn't mention anything about annual inspections of the hub. So we asked them that they would need to put some inspection criteria for the hub. Basically, that was our review for the flywheel. Any questions?

(No response.)

MR. HONCHARIK: Thank you.

MS. UMANA: Okay. We're going to move on to Greg Makar. He's going to present on the steam generator program.

MR. MAKAR: The review in Sections 5.4.2.1 and 5.4.2.2 address a variety of materials and design and inspection topics. Our slide here just refers to the one open item that we had, and that is a -- the accident induced leakage performance criterion in the tech specs, which was the value proposed is an acceptable value, but that's also described in a number of places in the tech spec bases.

We found some potential inconsistencies

and weren't certain that it, that the value that we're using was consistent with the accident analyses in Chapter 15. So after discussing that with them in our radiation protection and accident consequences branch, we found that it was really an administrative issue, that there just needed to be some, some edits in the tech spec bases to --

MEMBER STETKAR: That's what I wanted to make sure. I thought that -- I'm glad you clarified that. So it's simply making sure that the citations of things like .3 GPM and .6 GPM in the tech spec bases align with the values that were used for transients and steam line breaks respectively in the Chapter 15 analyses, right. You're not questioning it all, the 150 gallons per day.

MR. MAKAR: Operationally.

MEMBER STETKAR: Operational limit at all. Okay, thanks. Thank you.

MR. MAKAR: So the applicant has since responded to that and clarified that. So we are -- we'll be closing that open item and tracking it as confirmatory.

MS. UMANA: Okay, and just to top off the presentation I provided.

MEMBER STETKAR: Don't go too far yet.

MS. UMANA: Oh.

MEMBER STETKAR: Chris, no Chris.

MS. UMANA: Oh.

MEMBER STETKAR: I know where he's going. I'm trying to turn him off for a second.

MS. UMANA: This is just a courtesy list of the acronyms used throughout Chapter 5 and some that you probably heard in the presentation, if you are interested in referencing it.

MEMBER STETKAR: As usual, they can't shut me up. They try.

CHAIRMAN BALLINGER: We don't want to.

MEMBER STETKAR: Yes, you do but --

CHAIRMAN BALLINGER: Okay, we do.

MEMBER STETKAR: And I'm okay with that. Yesterday, I noted that the applicant has included this Appendix 5A specifically addressing the inter-system LOCA evaluations, and I had not seen in other design certifications such an extensive evaluation of that topic, and I'd like --

Again, I think that's very, very good. It shows a lot of thoughtfulness. I had a question for the staff, and this is -- I saved it for you because it's I think more pertinent for you.

In their evaluations, in particular of

the shutdown cooling system, the safety injection system and containment spray system, they rely substantially, I don't think entirely but substantially, at least in the DCD, on the fact that they designed those piping systems outside the containment to a pressure of 900 pounds, 900 psig, which they say is 40 percent of the normal reactor coolant system operating pressure of 2,250 psi.

I think that's a good idea, you know. They've in fact enhanced the design of those systems. The reason I have a question for the staff, and that I'd like to have a little discussion about this is that they repeatedly refer to things like this is in accordance with the NRC's position presented in the NRC Letter Reference 2.

And the Letter Reference 2 is the NRC letter preliminary evaluation of the resolution of the inter-system loss of coolant accident issue for the Advanced Boiling Water Reactor design pressure for low pressure systems, Docket No. 52-001. So we've got that on the record.

The staff's SER says well, you looked at that 900 pounds and you concluded that it's consistent with past applicant practices that you found acceptable, and you have reasonable assurance that at this design pressure, the integrity of the SES will be maintained.

My question is I haven't seen that argument used in any other design certification, and this seems to be establishing a precedent, such that if I just come in and claim that my shutdown cooling system, residual heat removal system or any other low pressure system at 40 percent of my reactor coolant system operating pressure, I get carte blanche. I can't have an interfacing system LOCA, because that is NRC's position.

And in this particular application, I can personally point to other things that provide additional protection in those systems against interfacing system LOCAs, and therefore draw my own conclusion about the susceptibility of those systems to breaks or the likelihood of their being overpressurized and then failing.

But what I'm asking the staff is is this an NRC regulatory position, because as we are all painfully aware, that once this is written in a safety evaluation report for this applicant, it will become regulatory precedent. I think there's a danger in doing that. I think that use of enhanced piping system design is part of the -- part of the solution.

I think that judicious locations of pressure relief valves, temperature indications, pressure indications, automatic or manual isolation release flow paths and those types of things are all -- testing. For example, how frequently do you go in and test interstitial spaces, to make sure that they're not overpressurized, that you don't have leakage that you perhaps couldn't monitor.

They're all part of the solution of this, not just simply saying well I've designed my piping to greater than 40 percent of reactor coolant system pressure. So I'd ask you to take that and come back and say is that a formal NRC regulatory position regarding interfacing system LOCAs.

The reason I'm concerned about that is twofold. Number one, from just a regulatory footprint, and number two, I certainly don't want to in the future go fighting battles with people doing risk assessments, saying that we don't need to consider thinking about interfacing system LOCA because the NRC staff has told us that we are absolutely safe because our piping is designed to 40 percent of reactor coolant system pressure. We've faced those battles in the past.

So the other thing I'll bring up is that the -- in research, they're currently working on the Level 3 PRA project, and as part of that project, they've had rather extensive work done on interfacing system LOCAs. I don't know what the current status of that is. We had a briefing some time recently, which might have been several months ago.

As part of that process, there was a report produced out at PNNL, and I'll give you the citation. It's PNNL 24783, Expert Elicitation to Support Interfacing System Loss of Coolant Accident Modeling, and the Level 3 PRA project is not using that report verbatim, but they're using information from that report to inform their treatment of interfacing system LOCAs.

I'm curious whether this old letter report, and I trace back a little bit on the history of the thing with NUREG/CRs, not NUREGs. NUREG/CRs, some of which I can't even find in ADAMS because they're so old, whether that is still consistent with the NRC regulatory position, research position and so forth on susceptibility to interfacing system LOCAs.

So if you can kind of take that kind of

long rambling approach. I'm not necessarily -- I'm not right now, because this is a subcommittee meeting, I personally am not questioning the applicant's conclusions regarding interfacing system LOCA.

But I base my conclusion not only on that 900 pounds. I base it on other features of the system's designs that I've looked at and what they claim they will be doing in terms of monitoring and testing and inspection and so forth.

So and I, you know, it's just I had to bring it up as I did yesterday because neither the applicant nor the staff made any mention in their presentations about, you know, Appendix 5A, about this analysis. Yet in the SER, you've addressed it and the applicant obviously has done quite a bit of work, time working on this.

If we're design certification, they're beyond design basis events. I will admit that. They can be really, really important if not addressed sufficiently in the world of risk assessment, as you're well aware, and now I'm done sir.

CHAIRMAN BALLINGER: Thank you. Other questions from members at this point? We're getting the phone line open. We should hear the eruption of crackles shortly. While we're doing that, are there any folks in the audience that would like to make a comment or a statement?

(No response.)

CHAIRMAN BALLINGER: Yes, it is. It's open. Is there anybody out there on the line? Can you identify yourself or make yourself known so we can verify that this line is open? Maitri's not even up.

MS. BANNERJEE: Yeah. This is Maitri.

CHAIRMAN BALLINGER: She is up. Thank you. Now is there anybody else out there that would like to make a statement?

(No response.)

CHAIRMAN BALLINGER: Hearing none, I think we're done there. Okay. Now in closing, can we go -- we'll go around the table, start with Pete this time.

MEMBER RICCARDELLA: I have no comments.

CHAIRMAN BALLINGER: Dana?

MEMBER POWERS: (No audible response.)

CHAIRMAN BALLINGER: Matt?

MEMBER SUNSERI: No comments, thanks.

MEMBER STETKAR: Nothing more, other

than to say I was pretty impressed with the staff's review of Chapter 5 in particular. Did a pretty, a pretty doggone thorough review. So good job.

CHAIRMAN BALLINGER: Jose?

MEMBER MARCH-LEUBA: No comment.

CHAIRMAN BALLINGER: Charles, Charlie?

MEMBER BROWN: No more comments.

CHAIRMAN BALLINGER: You're not even here. Oh, there you are over there.

MEMBER BROWN: I am.

CHAIRMAN BALLINGER: Joy, I don't know where her name tag is.

MEMBER REMPE: It's hiding among the electrical cords. But I have no additional comments. Thanks for your efforts and your presentations today. Thank you for your time.

CHAIRMAN BALLINGER: Well, we thank you very much. Again, I would like to express my gratitude for the time and effort and the thoroughness of your presentations, both the applicant and the staff, and with that, we are adjourned.

(Whereupon, the above-entitled matter went off the record at 9:31 a.m.)

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Presentation to the ACRS Subcommittee

Korea Hydro Nuclear Power Co., Ltd (KHNP)
APR1400 Design Certification Application Review

Safety Evaluation with Open Items: Chapter 5

REACTOR COOLANT SYSTEM AND CONNECTING SYSTEMS

SEPTEMBER 22, 2016

- **Technical Staff Presenters**
  - John Budzynski – DCD Sections 5.2.2, 5.4.1.2, 5.4.7, & 5.4.1.12
  - Dan Widrevitz – DCD Section 5.2.3
  - Jessica Umaña – DCD Section 5.2.5
  - John Honcharik – DCD Section 5.4.1.1
  - Greg Makar – DCD Sections 5.4.2.2
- **Project Managers**
  - Jeff Ciocco – Lead Project Manager
  - Jessica Umaña – Chapter 5 Project Manager

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- **Gregory Makar**
  Materials & Chemical Engineering Branch
- **Steven Downey**
  Materials & Chemical Engineering Branch
- **Eric Reichelt**
  Materials & Chemical Engineering Branch

Overview of Design Certification Application, Chapter 5

SRP Section/Application Section		No. of Questions	Number of OI
5.2.1.1	Compliance with the Codes and Standards Rule, 10 CFR 5055a	3	0
5.2.1.2	Compliance with Applicable Code Cases	2	0
5.2.2	Overpressure Protection	7	3
5.2.3	Reactor Coolant Pressure Boundary Materials	22	1
5.2.4	Inservice Inspection and Testing of the Reactor Coolant Pressure	2	0
5.2.5	Reactor Coolant Pressure Boundary Leakage Detection	3	0

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Overview of Design Certification Application, Chapter 5

SRP Section/Application Section		No. of Questions	Number of OI
5.3.1	Reactor Vessel Materials	0	0
5.3.2	Pressure-Temperature Limits, Pressurized Thermal Shock, and Upper-Shelf Energy Data and Analyses	1	0
5.3.3	Reactor Vessel Integrity	0	0
5.4	Reactor Coolant System Component and Subsystem Design	2	1
5.4.1.1	Reactor Coolant Pump Flywheel Integrity	12	6
5.4.1.2	Reactor Coolant Pump	0	0

Overview of Design Certification Application, Chapter 5

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SRP Section/Application Section		No. of Questions	Number of OI
5.4.2.1	Steam Generator Materials	3	0
5.4.2.2	Steam Generator Program	6	1
5.4.3	Reactor Coolant Piping	0	0
5.4.7	Residual Heat Removal System	4	1
5.4.10	Pressurizer	0	0

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Overview of Design Certification Application, Chapter 5

SRP Section/Application Section		No. of Questions	Number of OI
5.4.11	Pressurizer Relief Tank	0	0
5.4.12	Reactor Coolant System High-Point Vents	10	0
Totals		77	13

Technical Topics
Section 5.2.2 – Overpressure Protection

Technical Topics

- System Design
  - Overpressure protection for (1) RCS, (2) primary side of systems connected to RCS, and (3) secondary side of SGs

    - RCPB overpressure protection consist of 4 POSRV, 2 SCS RV, and 20 MSSV (5 per steamline)
  - SCS suction line relief valves provide sufficient pressure relief capacity to mitigate the most limiting low temperature overpressure protection (LTOP)

Technical Topics
Section 5.2.2 – Overpressure Protection

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<u>Open Items</u>

- **RAI 8244, Question 05.02.02-1** – Provide information regarding the analysis performed to demonstrate that POSRV and SCS RV provide sufficient capacity to maintain the pressure below the limit during full power and LTOP conditions.
    - **Status:** The response was received on 12/11/2015. The POSRV analysis was adequately addressed; however, the LTOP analysis did not address staff concerns related to methodology, codes, input parameters, and assumptions used in the LTOP analysis. This question is closed unresolved.
- **RAI 8609, Question 05.02.02-7 Follow up question** - Provide a description of the analysis methodology, computer codes, input parameters, and assumptions used in the analysis of the limiting LTOP event(s). Provide justification that the inputs are suitably conservative.
    - **Status**: The response was received on 6/9/2016. The follow up RAI response is currently in evaluation.
- **RAI 8244, Question 05.02.02-5 -** Address the difference between the pre-service testing requirements and Chapter 14 test.
    - **Status**: The response was received on 6/9/2016. The response was found acceptable because the complete pre-service testing is included in the Section 14.2 rewrite. The staff confirmed the revision. This question is closed resolved.

Technical Topics
Section 5.4.1.2 – Reactor Coolant Pumps

Technical Topics

- Pump Design
  - RCP is a vertical shaft, single-stage, centrifugal pump
  - Loss of Component Cooling Water Event

    - CCW flows to pump/motor bearing oil coolers, motor air coolers, and high-pressure seal water cooler stopped

    - CVCS continues to provide seal water injection flow to the RCP seals

    - Pump/motor bearings designed to withstand loss of CCW for 30 minutes

    - CCWS Division 1 supports charging pump mini flow heat-exchanger in CVCS and all coolers on the RCPs
  - RCP seal design is improved to have low seal leakage rates during a station black-out (SBO) event

Technical Topics
Section 5.4.1.2 – Reactor Coolant Pumps

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Open Items

- **RAI 8555, Question 05.04-2, Part 1** – With respect to Loss of Component Cooling Water, provide a response to clarify whether the loss of CCW Division 1 affects the charging pump operation during the 30-minute period following the event
  - **Status:** The response was received 4/29/2016. The response was found acceptable because of the loss of the CCW to the charging pump only has a long term effect on its operation. This question is closed resolved.
- **RAI 8555, Question 05.04-2, Part 2** – Revise DCD Section 5.4 to include a summary of the seal leakage test results that show the low seal leakage rate during a simulated SBO event.
  - **Status:** The response was received on 4/29/2016). The response was found acceptable because the DCD is being revised to include a reference to the seal test report. This question is closed resolved.

Technical Topics
Section 5.4.7 – Shutdown Cooling

Technical Topics

- System Design
  - Configuration Similar to Current PWR Designs,
    - Two independent trains – SCP interchangeable with CSP
  - Used during startup/shutdown and refueling operations
    - During shutdown and refueling RHR is placed in service below approximately 450 psig and 350°F
- Evaluated against GL-2008-01 and RIS 2013-09
- Evaluated against NEI 09-10 Rev 1a-A – Prevention and Management of System Gas Accumulation

Technical Topics

Section 5.4.7 – Shutdown Cooling

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Open Items

- **RAI 8614, Question 05.04.07-4** – Provide ITAAC to the SCS design that addresses GL-2008-01 and NEI 09-10 with respect to managing gas accumulation during plant operations
  - **Status:** The response was received on 7/8/2016. The ITAAC was adequately addressed and in compliance with NEI 09-10 Revision 1a-A; however, NEI 09-10 was referenced in the DCD. The staff will discuss the need to include the reference with the applicant. This question currently remains in evaluation.

Technical Topics
Section 5.4.12 – RCS High Point Vents

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Technical Topics

- 10 CFR 50.34(f)(2)(vi) and TMI Action Plan Item II.B.1 require high point vents, and 10 CFR 50.46a specifies acceptance criteria
  - Remove non-condensable gases from high points in the RCS to maintain adequate core cooling (enhance natural circulation)
  - The RCS high points are the reactor vessel head, pressurizer, and steam generator U-tubes; however, individual U-tubes are not required to have high point vents (10 CFR 50.46a)
- APR1400 High Point Vent Design: Standard Configuration
  - Safety-grade vent paths: (1) reactor vessel head to in-containment refueling water storage tank (IRWST); and (2) pressurizer head to IRWST
  - Each safety-grade vent path has two parallel vent paths with two normally closed solenoid-operated valves in series
  - Small-break loss-of-coolant accident analyses bound potential vent line breaks upstream of isolation valves
- Operating procedures described at a high level in the DCD; detailed operating procedures to be developed by COL applicant

Technical Topics

Section 5.2.3 – Reactor Coolant Pressure Boundary

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Open Item

- **Open Item, DCD Clarification** - It was determined that a portion of DCD text providing additional detail beyond the staff criteria for establishing that austenitic materials used for the RCPB will not be sensitized should be removed from the DCD. Since the DCD will be granted finality by rulemaking, this removal is necessary as the staff could not make a specific regulatory finding regarding its accuracy.
  - **Status:** The applicant has been notified and the staff has scheduled a meeting with the applicant to review this.

Technical Topics

Section 5.2.5 – Reactor Coolant Pressure Boundary Leakage Detection

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<u>Technical Topics</u>

- Regulatory Requirements: GDC30, GDC2
- Review Guidance: SRP Section 5.2.5
  For GDC 30 on RCPB leakage detection, the review is based on RG 1.45.
  For GDC 2 on seismic design, the review is based on RG 1.29 and RG 1.45.
- Review Areas: leakage detection capability, sensitivity, and response time; leakage detection systems; seismic qualification; leakage instrumentation in the control room; prolonged low-level RCS leakage; separation of identified and unidentified leakage; intersystem leakage; plant TSs; initial testing program; ITAAC; and COL information items.
- Three RAIs were issued and all are being resolved pending confirmation in the next revision of the DCD.
- One COL information item.

Technical Topics
Section 5.4.1.1 – Reactor Coolant Pump (RCP) Flywheel Integrity

Technical Topics

RCP flywheel analyzed to prevent fracture and possible missile.

- KHNP provided:
  - Material selection, fabrication techniques, preservice and inservice inspections and overspeed testing per NUREG-0800
  - Analysis per RG 1.14 (APR1400-A-M-NR-14001)
    - Critical speeds for ductile and non-ductile fracture
    - Fatigue crack growth

Technical Topics

Section 5.4.1.1 – RCP Flywheel Integrity

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Open Items

- **RAI 8641, Question 05.04.04.01-7** – The applicant needs to provide the method for determining fracture toughness.
    - **Status:** This question is currently waiting a response.
- **RAI 8641, Question 05.04.01.01-8** - The applicant needs to provide operating experience for the flywheel material 26 NiCrMoV 14-5.
    - **Status:** This question is currently waiting a response.
- **RAI 8641, Question 05.04.01.01-9** - The applicant needs to provide analysis using design acceptance criteria of 1/3 yield strength or provide justification for using 1/3 ultimate strength as the design acceptance criteria.
    - **Status:** This question is currently waiting a response.
- **RAI 8641, Question 05.04.01.01-10** - The applicant needs to provide analysis of the hub and why crack growth rates used for the flywheel are applicable to the specific flywheel material.
    - **Status:** This question is currently waiting a response.
- **RAI 8641, Question 05.04.01.01-11** – The applicant needs to provide in the DCD the maximum flaw size used in the analysis and that can be detected by the applicable NDE method.
    - **Status:** This question is currently waiting a response.

Technical Topics
Section 5.4.1.1 – Reactor Coolant Pump (RCP) Flywheel Integrity

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Open Items (continued)

- **RAI 8641, Question 05.04.01.01-12** - The applicant needs to include in the DCD the applicable PSI and ISI for the hub that is consistent with the flywheel to ensure the integrity of the hub is maintained.
  - **Status:** This question is currently waiting a response.

Technical Topics
Section 5.4.2.2 – Steam Generator Program

Open Items

- **RAI 494-8620, Question 05.04.02.02-6** - The assumed values for primary-to-secondary leakage described in the TS Bases (B 3.4.12 and B 3.4.17) appeared to be inconsistent with the accident analyses described in Chapter 15. Question 05.04.02.02-6 requested clarification of the leakage values in the TS Bases and how they are consistent with applicant's accident analyses and the Standard Technical Specifications (STS). Consistency between the accident analyses, APR1400 TS, and STS is needed to establish and implement a Steam Generator Program to ensure that tube integrity is maintained. This is part of meeting GDC 32.
  - **Status:** The response was received 7/26/2016. The response clarified the accident analysis assumptions and proposed revisions to TS Bases Sections B 3.4.12 and B 3.4.17 that are consistent with the accident analyses and STS. Therefore, the response is acceptable, this open item will be **closed,** and Question 5.04.02.02-6 will be tracked as a confirmatory item.

ACRONYMS

10 CFR – Title 10 of the Code of Federal Regulations
CCW – component cooling water
CVCS – chemical and volume control system
COL – combined license
DV – depressurization valve
FSAR – Final Safety Analysis Report
MCR – main control room
MOV – motor operated valve
PRT- pressurizer relief tank
PTLR – Pressure – Temperature Limits Report
PTS – pressurized thermal shock
PWR – pressurized water reactor
RAI – request for additional information
RCP – reactor coolant pump
RCPB – reactor coolant pressure boundary
RCS – reactor coolant system
RCSHPV – reactor coolant system high point vent
RG – Regulatory Guide
RHR – residual heat removal
RVHVS – reactor vessel head vent system
RWSP – refueling water storage pit
SDV – safety depressurization valve
SER – safety evaluation report
SRP – Standard Review Plan
SRV – safety relief valve