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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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APR1400 SUBCOMMITTEE

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WEDNESDAY

FEBRUARY 8, 2017

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ROCKVILLE, MARYLAND

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The Subcommittee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B1, 11545 Rockville Pike, at 8:31 a.m., Ronald G. Ballinger, Chairman, presiding.

COMMITTEE MEMBERS:

RONALD G. BALLINGER, Chairman

MARGARET CHU, Member

MICHAEL L. CORRADINI, Member

WALTER L. KIRCHNER, Member

JOSE A. MARCH-LEUBA, Member

DANA A. POWERS, Member

HAROLD B. RAY, Member

JOY REMPE, Member

PETER C. RICCARDELLA, Member

GORDON R. SKILLMAN, Member

JOHN W. STETKAR, Member

MATTHEW W. SUNSERI, Member

ACRS CONSULTANT:

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

CHRISTOPHER BROWN

ALSO PRESENT:

TONY AHN, KHNP

JOE ASHCRAFT, NRO

SUD BASU, Public Participant

ANDREW BIELEN, RES

ALEXANDRA BURJA, NRO

CHANG SOK CHO, KNF

WOOCHONG CHON, KEPCO NF

JEFF CIOCCO, NRO

MANSEOK DO, KEPCO NF

JAMES GILMER, NRO

SYED HAIDER, NRO

JOHN HONCHARIK, NRO

JANG, KEPCO E&C

REBECCA KARAS, NRO

HUNG JIN KIM, KEPCO NF

II KYU KIM, KEPCO NF

JONG SOO KIM, KEPCO E&C

KANGHOON KIM, KNF

TAE HAN KIM, KEPCO

YUNG HO KIM, KHNP

SHANLAI LU, NRO

DAE HEON LIM, KEPCO E&C

MICHAEL MCCOPPIN, NRO

MATTHEW A. MITCHELL, NRO

KEN MOTT, NRO

JIYONG OH, KHNP

MARK ORR, RES

SUNG KEW PARK, KEPCO NF

TAE SUN RO, KEPCO E&C

JAMES ROSS, AECOM

ROBERT SISK, Westinghouse

JEONG KWAN SUN, KHNP

YIXING SUNG, Westinghouse

CARL THURSTON, NRO

CHRISTOPHER VAN WERT, NRO

ANDREA D. VEIL, Executive Director, ACRS

DANIEL WIDREVITZ, NRO

GEORGE WUNDER, NRO

PETER YARSKY, RES

TAE YOUNG YOON, KEPCO NF

*Present via telephone

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P R O C E E D I N G S

8:31 a.m.

CHAIRMAN BALLINGER: The meeting will now come to order.

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

I'm Ron Ballinger, Chairman of the APR1400 Subcommittee.

ACRS members present are Joy Rempe, Jose March-Leuba, Walt Kirchner, Pete Riccardella, Mat Sunseri, Dana Powers, Dick Skillman, Harold Ray, Margaret Chu and our consultant Stephen Schultz, former ACRS member.

I believe we're going to be joined by Mike Corradini and Charles Brown, I think.

The purpose of today's meeting is for the subcommittee to receive briefings from Korea Hydro and Nuclear Power Company, KHNP, regarding their designs verification application and the NRC staff regarding their review of the safety evaluation specific to Chapters 4, The Reactor.

This meeting is the sixth in a series of meetings of our subcommittee to review KHNP application and related NRC staff safety evaluations.

The rules for participation in today's meeting were announced in the Federal Register on February 7th, 2017.

The meeting was announced as open, but, portions can be closed as needed. I don't think that's going to be the case, but to protect proprietary information to KHNP or its vendors pursuant to 5 USC 552(b)(c)(4).

No requests for making a statement to the subcommittee has been received from the public.

A transcript of the meeting is being kept and will be made available as stated in the Federal Register Notice. Therefore, we request that participants in this meeting use the microphones. And, there's a little button on the lower part and the little light has to turn green to make it work.

The bridge -- a bridge number and a password were published in the Federal -- on the NRC website, excuse me. To minimize disturbance, the public line will be kept in a listen only mode. And, those of you on the public line, if you would be kind enough to mute your line when you're not participating because there's feedback and things.

The public will have an opportunity to

make a statement or provide comments as designated towards the end of this meeting.

There is an additional line that the staff has requested for NRO staff and contractors to participate in certain portions of their presentations.

Please let Chris know when this line should be opened. And, Chris Brown is the Designated Federal Official for this meeting.

I'll request that the attendees and participants silence their cell phones and other electronic devices please.

Now, I invite Jeff Ciocco, there he is, NRO Project Manager to introduce others and start the briefing.

MR. CIOCCO: Yes, good morning. My name is Jeff Ciocco. I'm the Lead Project Manager for the APR1400 design certification. Thank you for having us today for the APR1400 Chapter 4 Safety Evaluation with Open Items.

And, with that, let's get on with it.

MR. SISK: Rob Sisk, Westinghouse representing the KHNP and the APR1400 design. We look forward to continuing our discussion with this ACRS to present the APR1400 DCA today in Chapter 4.

So, I echo Jeff's comments, we're ready to go. Are you ready?

CHAIRMAN BALLINGER: Yes.

MR. SISK: Then I'll turn it over to Il Kyu Kim to lead us through Chapter 4.

MR. I. KIM: Good morning ladies and gentlemen. My name is Il Kyu Kim from KEPCO-ENF.

We are going to present this Chapter 4 reactor today.

This presentation consists of overview of Chapter 4, Section Summary, Summary and Attachments.

In the Section Summary, summary or each section will be presented from Section 4.1 to 4.6.

The Chapter 4 consists of six sections, a summary description, fuel system design, nuclear design, thermal-hydraulic design, reactor materials and functional design of reactivity control system.

I will present Section 4.1 and 4.2. Manseok Do will present Section 4.3; Kanghoon Kim will present Section 4.4; Jong Soo Kim will present Section 4.5 and 4.6.

This slide shows the summary to the documents with related sections. DCD to topical records and the six technical records are submitted for Chapter 4.

This table shows the summary of the APR1400 core and the fuel design. Core power is 3,983 megawatt to thermal. The fuel rod lattice is 16x16.

The number of fuel lattice in our fuel assembly are 236. The number of fuel assemblies in core are 241. The number of control element assemblies in core are 93.

The active fuel lengths is 150 inches. Maximum peaking factor is 2.43 and maximum fuel rod average burnup is 60,000 megawatt of metric ton uranium.

This tables shows the primary core used in this section -- used in each sections.

There is no open items for Section 4.1.

From this slide, let me give you a presentation for the Section 4.2.

This slide shows the PLUS7 design features. PLUS7 fuel assembly has been developed here for combustion engine type nuclear power plants. Therefore, PLUS7 has typical features of combustion engine type fuel, such as core spring for hold down spring and the big guided thimbles.

As I already explained it, a regular

fuel rod in assembly is 16x16, the numbers of fuel rods in assembly are 246 and the thermal core of cladding is ZIRLO.

The number of the material we tied in the back is Stainless Steel 304, except hold down spring of which material is Inconel 718.

One top, one bottom and the one protective grade are used at the top and bottom portion of the fuel assembly. The material with these grids are Inconel 718.

Nine mid grids and four guide thimbles are used and in the metal of both is ZIRLO.

This slide explains the irradiation experience of PLUS7 fuel. We had two surveillance programs for PLUS7 fuel.

One is with lead test program, the other one is commercial surveillance program.

Four lead test assemblies were loaded and examined here for commercial supply. And, the four assemblies were selected for per power examination among the fuel assemblies commercially supplied for Hanbit Unit 5, Cycle 5 which include these four assemblies as commercial surveillance assemblies.

Four site examinations were conducted

for lead test assemblies and the commercial surveillance assemblies.

Also, hot cell examination of related test assembly has been completed after irradiation.

Per site examination and hot cell examination results showed all design requirements were met.

Now, I explain about the operating experience. About 5,000 PLUS7 fuel assemblies have been supplied as of 2016.

MEMBER MARCH-LEUBA: Can I ask a question? I mean, each core loads 241 fuel assemblies, correct?

MR. I. KIM: Yes.

MEMBER MARCH-LEUBA: So, 5,000 is 24 cores, roughly?

MR. I. KIM: Twenty-four core --

MEMBER MARCH-LEUBA: Five thousand divided by 241.

MR. I. KIM: Five thousand -- yes, approximately.

MEMBER MARCH-LEUBA: Approximately there are 20. Yet, you're still talking about four LTAs, so --

MR. I. KIM: Pardon me?

MEMBER MARCH-LEUBA: You're talking about four LTAs. So, it seems to me that only four elements have been loaded in the core. And what's this 5,000?

MR. I. KIM: Actually, the LTA loaded in 26 --

MEMBER MARCH-LEUBA: Okay.

MR. I. KIM: -- and the five short ones. Anyway, after that, we supply the fuel assembly commercially of about 20 years for operating power plant.

MEMBER MARCH-LEUBA: For the last 20 years, you're been supplying or --

MR. I. KIM: Already we supplied about --

MR. SISK: Perhaps, I can just quickly.

The PLUS7 fuel was developed back around -- in the 2000 as you could say. And, the LTA, this is kind of a summary history were done in 2006. It's the fuel PLUS7 is being used in Korea since that time.

So, 2006 to now we're at 2017, PLUS7 fuel has quite a bit of experience.

MEMBER MARCH-LEUBA: But for the LTAs can't be relevant because you have 5,000 of the other ones?

MR. SISK: Except those were the ones where you did the actual hot cell exams and so, that's part of the history of the development of the PLUS7.

CHAIRMAN BALLINGER: What's the peak burnup?

MR. I. KIM: Peak burnup, I'm no -- I don't remember quite what --

MR. DO: The peak burnup is 50 gigawatt a day for material uranium.

CHAIRMAN BALLINGER: Well, okay, is that the -- that's the design peak burnup. But, what's the actual peak burnup been for these 5,000 PLUS7 fuel assemblies?

MR. I. KIM: I'm not sure.

CHAIRMAN BALLINGER: I couldn't find it.

MR. CHON: Yes, that's about 59,500 burnup.

CHAIRMAN BALLINGER: Oh, okay. Okay, that'll do.

MR. CHON: This is Woochong Chon from KEPCO Nuclear Fuel.

CHAIRMAN BALLINGER: Thank you.

I should ask one more question. What's

the failure rate been?

MR. I. KIM: Failure rate?

CHAIRMAN BALLINGER: Yes, how many fuel rods --

MR. I. KIM: Ten fuel rods have been failed.

CHAIRMAN BALLINGER: Okay.

MR. I. KIM: Okay.

CHAIRMAN BALLINGER: Thank you.

MR. SCHULTZ: I have a question related to the fuel failure. You mention in the fuel topical report that you had addressed fretting failures for this fuel design very well. You haven't had any experience with fretting failures for many years now, many years.

When you say ten -- you've had ten fuel failures, fuel pin failures, what have they been due to? What have those failures been due to? Have you investigated the cause of those failures?

MR. I. KIM: Seven failed rods, the cause of the failure of seven fuel rods is debris failure.

MR. SCHULTZ: That was debris?

MR. I. KIM: Yes. And --

MR. SCHULTZ: And, so, when did that

happen? Was that many -- several years ago? You've got some debris barriers now in your fuel design system.

MR. I. KIM: I don't remember the date when the failure happened. And, I have no information now.

CHAIRMAN BALLINGER: So, to close the loop, seven debris failures. What were the other three?

MR. I. KIM: One is unknown, two is manufacturing failure.

CHAIRMAN BALLINGER: Thank you.

MR. SCHULTZ: Thank you.

MR. I. KIM: You are welcome.

Okay, now, let me continue with the presentation.

Single Unit 3 which is the referenced plant of APR1400 state 30 commercials operating with PLUS7 fuel assembly on December 20, last year.

PLUS7 fuel assemblies will be supplied for 4 Barakah APR1400 nuclear power plants in UAE and five APR1400 nuclear power plants in Korea.

The design requirements of Section 4.2 are 10 CFR Part 50 Appendix A and 10 CFR Part 50.46. And the NRC guidance are Regulatory Guide 1.206, Section 4.2 and the Standard Review Plans Section 4.2.

Design criteria related to fuel system design are listed in this table. All the evaluation for this design criteria were performed and described in Section 4.2.

And, the results show that each requirements are met.

This slide explains about the open items. Open items in the Section 4.2 is related to PLUS7 fuel assemblies structural analysis.

There are nine RAIs which is not responded yet on the PLUS7 seismic technical report. Fuel assembly tests and the seismic analysis have been performed to answer the RAIs.

All of the tests have been performed except the damping test.

RAIs will be responded by February 28th, the end of this month and the technical report will be revised.

MR. SCHULTZ: I have a question.

MR. I. KIM: Yes?

MR. SCHULTZ: In the topical report for the fuel design, you have listed that this area would be site specific or because of seismic.

MR. I. KIM: Yes.

MR. SCHULTZ: Yes.

MR. SCHULTZ: Because of seismic, is that still a condition of the discussion with the staff that there would be a site specific evaluation of the seismic loading or have you come to an approach where it is not site specific? You've bounded it in some fashion?

MR. I. KIM: Yes, actually, the topical report does not include that seismic LOCA evaluation, but DCD includes the seismic LOCA calculation. Yes, it's listed in the technical report that's listed in this slide. This technical report describes the analysis results for the seismic LOCA analysis.

MR. SCHULTZ: And, what does that assume for the seismic loading? Is there -- is that a bounding evaluation or will there be a site specific seismic evaluation in addition to what you're doing here?

MR. SISK: Rob Sisk, Westinghouse.

In general, as you know, there is Chapter 3 has yet to be reviewed. We'll talk about these 37, 38, the seismic requirements. The APR1400 is a .3G plant. The fuel is being evaluated and demonstrated to be acceptable for the plant design of .3G.

MR. SCHULTZ: Good, thank you. Go ahead.

MEMBER CORRADINI: Just to follow up, so then, that's the envelop and then if a particular site exceeds the envelop, then it has to be site specific?

MR. SISK: Of course.

MEMBER CORRADINI: Okay, that's fine.

MEMBER REMPE: So, I have a question to follow on earlier about the maximum burnup of the PLUS7 fuel.

In the report about the effect of thermal conductivity degradation, it talks about the FATE code, FATE 3, is it FATE 3S3B, I don't know how you refer to it, that's another acronym said.

But, anyway, it said it's only been calibrated up to maximum burnup greater than -- or up to 23,000 megawatt days per metric ton uranium.

And, then, I know there's been some discussion about what's going on about thermal conductivity degradation and how you looked at the results from your analysis versus what would happen if you went to a higher burnup.

And, could you talk a little bit about what your plan is to address some of the staff questions about thermal conductivity degradation and how that's progressing? Because that does impact what we're reviewing here on the fuel today.

MR. CHON: This is Woochong Chon from KEPCO Nuclear Fuel.

We are still discussing about that issue which is DCD with analysis test. And, we added more housing data to justify our results. The added housing data contains more higher burnup which is almost the same as a range of PLUS7 fuel.

MEMBER REMPE: Okay.

MR. CHON: But, we will conclude DCD issue maybe within this month.

MEMBER REMPE: Okay, thank you.

MR. DO: Good morning, everyone. I am Manseok Do from KEPCO Nuclear Fuel. I am in charge of the nuclear design for APR1400.

Today, I am going to make a brief description about Chapter 4.3 of the APR1400 DCD.

And, I'll start with general characteristics of load patterns and fuel assemblies for APR1400 core.

The loading pattern on APR1400 core

consists of three batch scheme. It reduces the peaking factor.

The core that time is based on refueling to of approximately 18 months with replacement of one over three which is including each refueling outage.

As you can see in this picture, the fuel assembly is, again, a system by 16x16 containing 236 pure or one over equation rods with control rod guide two and one instrumentation guide two.

The fuel rod contain uranium dioxide pellets are various in each one. And, the rods contain gadolinia-urania pellets.

The system improves the long term control of the LOCA assembly power distribution.

If the ratio number of fuel assemblies or fuel rods in each month, a number of gadolinia rod assembly and gadolinia contents for each fuel assembly type.

MEMBER SKILLMAN: Before you change that image, let me ask this question. In the blue first carry, you indicate three batch system, refueling interval of 18 months.

But, the tech or the words that you spoke were approximately 18 months and the DCD at

4313 indicates approximately 18 months. Why have you used that word approximately?

MR. DO: The 18 months cycle contain over a period. So, you get operation time will be varied between 400 to 500 EFPDs. So, I mean by about, it will be expected 18 months for equilibrium cycle.

MEMBER SKILLMAN: Okay, thank you. I understand.

MR. SCHULTZ: With regard to the gadolinia fuel rods -- containing fuel rods, I didn't see it, have you demonstrated that in terms of thermal performance that the gadolinia rods are never limiting? The conductivity will be lower than for the UO2 rods, so, therefore, have you done analyses that demonstrate that gadolinia rods are not thermally limiting because of the burnable poison within them?

MR. DO: Do you mean if they --

MR. SCHULTZ: For the analysis that's performed in safety analysis either for operating performance -- we're not -- we're talking Chapter 4, not Chapter 15, but as you do your thermal performance analysis, I presume those rods, in terms of their power are not limiting and so the thermal -

- the melting -- the rod temperature, for example, is not a limiting -- cannot be the limiting rod even though the thermal conductivity in those rods will be somewhat lower than for UO2 rods.

MR. CHON: This is Woochong Chon from KEPCO Nuclear Fuel.

Generally, the gadolinia fuel has to do much less enrichment. So, most cases, gadolinia fuel is not limiting.

MR. SCHULTZ: Okay. Fine, thank you.

MR. CHON: No problem.

MEMBER MARCH-LEUBA: In the nuclear enrichment, is the enrichment actually --

MR. DO: There's a pass of fuel per 2.0 with percent.

MEMBER MARCH-LEUBA: Say again?

MR. DO: 2.0 with percent for top and bottoms.

So, these are basis for the nuclear design of the APR1400 are, as part of first the core power, the solutions are controlled through that the power distribution.

Second, when the reactive is noted and tends to compensate for rapid reactive decreases in the power plant.

Finally, control systems are capable of providing an acceptable margin and of controlling power decision.

But the force of the nuclear is to provide the limits under power distribution so that we don't damage the fuel during the power operation.

But, factor for full power condition is 2.43 which is based on the LOCA limit and the core average.

The minimum DNBR is 1.29 as described in Section 4.4. And, the maximum peak fuel rod is 50 given today on the uranium.

But, the limits on the power and the initial conditions described in Chapter 6 and 15.

MEMBER SUNSERI: I have a question on that slide. So, in a few places in the design certification document, it describes this plant as designed for base loaded operation but capable of load following.

So, in your definition of normal operation under the core power distribution, does that include load following operations and do these parameters bound the conditions that would be found during load following?

MR. DO: No, we don't intend to load

following operation.

MEMBER SUNSERI: You don't intend to load follow?

MR. DO: Just daily power or following top end load.

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

That word is corrected by the RAI Number 293-8332. Initially, we used the word power, but we corrected that word with power maneuvering. So, that means that our -- the APR1400, there is no intention to load follow operation anymore.

MEMBER SUNSERI: Okay, so, in various places like in Chapter 8 where it says the turbine generator is designed for -- or maybe it's 10, I can't remember, 10 -- for load following operations, that's been removed? Or changed to power maneuvering?

MR. OH: I think that the next revision, we probably we will change that inconsistent with Chapter 4. So, I assume that the usually turbine as it is, so that the initially, they had some function for the load following. Whether it's in a -- incorporate to the nuclear or the fuel.

So, that's the reason there's a genetic

function for the load following. But, in the fuel side, we don't have some function for the load following anymore.

So, even though it is some function for the load following on the turbine side, there's no function can be accomplished by the load following anymore.

MEMBER SUNSERI: Okay. So, maybe I understand this now a little better. So, the overall plant may be able to load follow, but the reactor's going to stay at full power because the turbine generator side is going to be designed to dump the steam or something so it can follow the load, but the reactor stays constant? No? Yes? No?

MR. OH: No, and for the fuel side, no load following operation.

MEMBER SUNSERI: Okay.

MR. OH: So --

MEMBER CORRADINI: So, can I ask Matt's question differently? Can you define the difference between load follow and power maneuvering? Is it just the rate of power change in the thermal power? I'm still not clear of the difference in the words.

MR. OH: Yes, this is Andy Oh, KHNP,

Western Office again.

For load following is for, you know, some constant operation for depending on a power level. When we use the power maneuvering means that we assume that both RPCS reactor power system or something if some -- then the power can be compacted to the some certain level of the power. Then we assume that is a power maneuvering operation. That's the reason we use the --

MEMBER CORRADINI: Yes, I understand. All right, did that help? I'm still a little unclear.

MEMBER SUNSERI: No, I think he's saying -- I think they're describing power maneuvering as just something that you would do to the plant to maneuver it and varying transients and whatever.

But, load follow is a, at least in this country, is defined as a capability to match the output of the plant to the demand of the system as it varies in significant ways, you know, like overnight or during the day or something of that nature.

And, typically, those load following operations, if the reactor is allowed to follow the load also causes power distribution concerns that need to managed. Right?

And, so, our questions are, we've seen in the DCD where the term load following has been used and we're wondering, you know, if it's been thoroughly analyzed for that.

And, what I hear you saying is the reactor is not intended to be operated in a load following manner, is that correct?

CHAIRMAN BALLINGER: So, let's just get the definitive statement because there's been a lot of confusion and I think it might have been confusion with respect to definitions and things like that as well.

MEMBER REMPE: But, apparently this --

CHAIRMAN BALLINGER: This plant is not designed to load follow, period.

MEMBER REMPE: But there has been an RAI that where the staff asked them to clarify it and you've committed to make that change.

MR. OH: Correct.

MEMBER REMPE: Can, maybe, again, there were a lot of RAIs, but can we see that RAI? Can it be sent to Chris, if we have it? Was it a recently -- I mean, this was brought up when we talked about Chapter 10 and it wasn't clear what was happening.

MEMBER SUNSERI: Andy gave that number. Could you repeat that number again so I can write it down? That's okay, we'll get it. We can get it later.

MR. OH: That is RAI Number 293-8332, question number four.

MEMBER SUNSERI: Okay, thank you.

MEMBER REMPE: So, out of curiosity then, in this Chapter, you still had load following. So, I guess it was submitted earlier and the staff was just silent about the load following. And, is that because they knew that you were going to remove it and that's why they didn't mention it? And, maybe that's a question for the staff.

MR. LU: Shanlai Lu from the staff.

Yes, I think load following, after we asked the question, we decided not to do the load follow in there. So, therefore, it's not an issue anymore from our perspective.

MEMBER REMPE: Okay, thank you.

MEMBER SUNSERI: Thank you.

MR. DO: This slide describes reactivity coefficients. The figure shows the dependence on the circulated temperature coefficient on the fuel temperature.

Also, at the beginning and at the end of the first cycles.

The fuel temperature coefficients are noted for the range of fuel temperatures throughout the cycle.

The mode of the temperature coefficient is slightly positive at power for a short time of the cycle.

But the net impact of the reactive equation is noted for whole power operating ranges throughout the cycle.

MEMBER MARCH-LEUBA: By burning up absorbers, you mean gadolinia rods, control rods?

MR. DO: Gadolinia rods.

MEMBER MARCH-LEUBA: The gad rods. So, that's built on the fuel not the coefficient? I mean that's --

MR. DO: I'm sorry?

MEMBER MARCH-LEUBA: That's built on the -- it's part of the coefficient?

MR. DO: Yes.

MEMBER MARCH-LEUBA: So, that is --

MR. DO: The gadolinia rods are the critical boron concentration is reduced. MEMBER MARCH-LEUBA: What -- say again?

MR. DO: Because they are gadolinia rods, we can reduce critical boron concentration at EOC. You know, MTC is dependent on much of it on. But it can be just above concentration, you can maintain MTC.

MEMBER MARCH-LEUBA: Okay, so the MTC is never positive. It would be using the gadolinia rods? Is the MTC post the MTC the moderator temperature coefficient positive with the gadolinia rods inserted?

MR. DO: For power about 3,000 there are -- uranium at the time MTC is slightly positive. But over reactive to coefficient where the fuel temperature was.

MEMBER MARCH-LEUBA: Okay, so the MTC is positive?

MR. DO: I think just at the time.

MEMBER MARCH-LEUBA: For a short period of time?

Yes, because the issue with positive MTC, the moderator temperature coefficient responds instantly to, at least, certainly to simultaneous to moderator heating. Whereas, the fuel has a two, three, five, ten second time response.

So, you may have -- I mean, whenever you have positive reactivity coefficients, you have to make sure that you don't have an accident that gets you a serious problem.

And, as I'm sitting here, I don't have any confidence that you have done an analysis. Have you done that analysis?

MR. SCHULTZ: Could you describe again, perhaps you mentioned it and I just missed the range, you indicated when you might begin to have a positive MTC, how long does it last? What's the burnup range in which you might have a positive MTC?

MR. DO: One kilowatt a day permitted to uranium.

MR. SCHULTZ: For that range? For a short range? Why does that happen just over a calculated short range? Is there a way to prevent that?

MEMBER MARCH-LEUBA: Well, what happens is you have too much reactivity and you had to put too much dissolved boron.

MR. SCHULTZ: Yes, I understand. That's a very short time frame.

MEMBER MARCH-LEUBA: Still, I mean, if it happens, it happens. I mean, you should --

MR. SCHULTZ: Is that actually occurring

in the cycle designs for other reactors with this fuel? The positive MTC? And, it's part of the control systems for those other reactors or is this something special about this reactor design?

MR. DO: Not special.

MR. SCHULTZ: Okay.

MR. DO: For extremely low power, at low power, MTC sometimes positive for other -- also other type of reactor.

MR. SCHULTZ: Low power? If you --

MR. DO: Not low power.

MR. SCHULTZ: Okay, okay. But, not near the very beginning of the cycle, that's controlled by the burnable poison. The positive -- it is not a positive MTC at beginning of cycle because of the burnable poison?

MR. DO: Not beginning of cycle.

MR. SCHULTZ: But sometimes later at hot zero power, it could be positive?

MR. DO: When the gadolinia depletions are made. When gadolinias are depleted.

MR. SCHULTZ: Yes.

MR. DO: The concentration right here at the time MTC is slightly positive at hot zero condition.

MR. SCHULTZ: Okay, okay, thank you.

MS. BURJA: This is Alex Burja from the staff.

I'd just like to add, again, so, there is a slightly positive MTC at hot zero power at beginning of cycle. But, during power operations, it becomes negative.

MEMBER MARCH-LEUBA: So, when you go to full temperature at a 100 percent power, everything is negative?

MS. BURJA: Correct.

MEMBER MARCH-LEUBA: So, the issue would be in startup? When your fuel is cold and you're relying on that cold fuel to heat up to give you that negative feedback?

I really would like to see some analysis because that is textbook program with fast transits.

MS. BURJA: Okay, I understand.

MEMBER MARCH-LEUBA: I mean, when your fuel is cold and your temperature coefficient on the liquid is positive, you must have a very large over shoot because you are at zero power, meaning you're not heating up the fuel.

So, you have to have a tremendous over shoot of power before you will heat up the fuel and get a negative feedback.

So, I will wait until you guys are here and I'll ask you what analysis you've done to confirm that you don't have startup problems.

MEMBER SUNSERI: But, I don't think it's uncommon for --

MEMBER MARCH-LEUBA: It's not uncommon but --

MEMBER SUNSERI: -- plants in the U.S. to have a slightly positive at the beginning of life right now.

MEMBER SKILLMAN: For every reload that we did for years, we had a positive coefficient for approximately the first week. And, as soon as we build in the fission product poisons, that moderator temperature coefficient became negative and remained negative. We were on 24-month cycles, but it was about the first 96 hours or so, the first four or five days.

MEMBER MARCH-LEUBA: You were just hoping you didn't have an accident those days?

MEMBER SKILLMAN: Well, no, no. We were controlling the reactivity to make sure that we did not have the accident that you are speaking about. We were very deliberate and we knew it was in the core operating limits report, just like the gentleman said. So, this is not new.

MEMBER MARCH-LEUBA: I'd just like to have the confidence that it has been analyzed.

MEMBER KIRCHNER: I think it has, if you read the detailed reports.

(OFF MICROPHONE COMMENTS)

MR. SCHULTZ: That is true. It did become an issue when the industry went to burnable poisons at these levels. But, it was addressed and so it has become, not a universal approach, but certainly one that is used by many BWRs.

MR. DO: The reactive control system of the APR1400 provides a shutdown margin considering single malfunctions over the reactivity control systems.

This figure shows the control element assembly pictures for core.

The APR1400 core is equipped with the 81 full strength CEAs and 12 partial strength CEAs. The internal materials in full strength CEA and partial strength CEA.

Therefore, 12 of the internal elements for full strength CEA and for elements for partial strength CEA.

The insertion limit is a function of the core power and the components to shutdown margin.

The final issue in this section is the stability. The APR1400 core is inherently stable to the total power because of the negative overall power correction.

Even though the general induce the power the stability shows oscillations may occur. The radial and azimuthal stability index is noted throughout the entire cycle.

If oscillation occurs, based on the core operating limits supervisory system, measurement of the disposition, the operator may move the core strength CEAs or partial strength CEAs to conserve any axial power oscillations and that they can be controlled effectively by partial strength CEAs in the figure 4.3 page 43 of DCD.

MEMBER MARCH-LEUBA: So this xenon oscillations will be mostly axial oscillations?

MR. DO: Yes, correct.

MEMBER MARCH-LEUBA: And, then, you have to sufficient feedback that they are not supposed to happen? That you have sufficient feedback, sufficient reactivity coefficient that will dampen them out under normal conditions? Is that not what your first bullet says? That oscillations will happen?

MR. SCHULTZ: First of all, is they're radial not axial.

MR. Y. KIM: This Yun Ho Kim.

Actually, the general oscillation temp out, so you don't have to worry about general oscillation.

MEMBER MARCH-LEUBA: You don't have axial oscillations because of --

MR. Y. KIM: Yes, actually, in term of radial oscillation, we have a monitoring part so they can be monitored.

And, in terms of axial, usually, have power. Usually, power creates a general oscillation damper.

MEMBER MARCH-LEUBA: So, you have performed some analysis that show that they damped out?

MR. Y. KIM: Yes, right.

MEMBER MARCH-LEUBA: And, if you were wrong and if it were to happen, the operator has access to control rods and he will know how to cancel them?

MR. Y. KIM: Yes, usually the

inherently, they are general oscillation is at -- they're small, small, small.

CHAIRMAN BALLINGER: One last thing about moderator temperature coefficient. Most of the PLUS7 fuel has been used in OPR1000 reactor?

MR. DO: Yes.

CHAIRMAN BALLINGER: So, what's the experience been in the OPR1000? Was that still -- it was a short period of time of burnup when you had a positive moderator temperature coefficient and that was handled easily? What's the experience been in OPR1000?

MR. DO: It's a similar --

CHAIRMAN BALLINGER: But, there's been no issue?

MR. DO: We set limiting conditions over operation for MTC. MTC is efficient for limiting at hot zero power.

CHAIRMAN BALLINGER: Okay.

MR. DO: And, that condition is used as an input.

CHAIRMAN BALLINGER: Okay.

MR. DO: We don't have it.

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

In addition to that, that is positive in moderate temperature. It's only a fear in initial core, not an equilibrium core.

CHAIRMAN BALLINGER: Okay, thank you.

MEMBER MARCH-LEUBA: Can you say that again? You were blocked.

MR. OH: Yes, for that positive MTC is only happen at initial core when it comes to equilibrium core, there's no positive MTC fears anymore.

MEMBER MARCH-LEUBA: Well, that's relevant information that you should have provided. So, that's very good. And, why is that? If anything, the initial core will have more gadolinia.

MR. OH: Yes, the more gadolinia we used in initial core, so as time goes, the gadolinia is burnable poison it's burned up and then the soluble poison is replaced the function of the burnable poison.

That's the reason for the moderate temperature coefficient has become slightly positive in initial core. But after that, when it come to the equilibrium core, we don't need to use much more burnable poison or something. That only happens in initial core.

MEMBER MARCH-LEUBA: Let me just follow up. In that initial core, do you have a problem with shutdown margin? I mean, what you're telling me is you have too much reactivity in the original core that is compensated by soluble boron?

Are you having a shutdown problem? A shutdown margin problem on the first core? Have you calculated the shutdown margin for the initial core?

If you're telling me that the MTC is probably positive on the initial core because the core -- that core has too much reactivity. And, I'm thinking, well, I need to shut it down, I need to have sufficient shutdown margin.

I assume you've calculated the shutdown margin for the initial core and it's okay.

MR. DO: It is okay. But, the MTC appears relatively to shutdown margin because when reactor temperature goes down because of MTC the reactor can rise. So, the reactor MTC affects the reactor for shutdown margin.

So, initial core shutdown margin is okay.

MEMBER MARCH-LEUBA: It is okay?

MR. DO: There are no open items for Section 4.3.

Thank you for your attention.

MR. SISK: I'm going to switch out these. But, if there's any questions for 4.1, 4.2, 4.3, we're going to switch out for 4.4, 4.5 and 4.6. It'll take just one minute.

MR. K. KIM: Okay, good morning. I am Kanghoon Kim from KEPCO Nuclear Fuel and I will present the DCD Section 4.4, Thermal Hydraulic Design of the APR1400.

The design basis of the DCD Section 4.4 are based on TDC 10 and TDC 12. Those are during the normal operations and AOOs, anticipated operational occurrences, the hot fuel load in the core will allow DNB.

It at least 95 probability at a 95 confidence level.

APR1400 special gadolinia limit is 1.29 with KCE-1 CHF correlation for PLUS7 closed design.

The temperature over uranium dioxide pellets to assure that no melting occurs.

MEMBER POWERS: On the gadolinia burnable poison rods, how does the melting temperature on that change?

MR. K. KIM: It then maybe slighter lower than uranium dioxide. I guess the melting temperature of the gadolinia pellets is approximately 2500 degrees.

MR. SCHULTZ: And, what you've listed here on the slide, just for clarity, is the -- it's the whatever you want to call it, the maximum fuel melting temperature for UO2, it's going to degrade with burnup over the course of the cycle.

So, and you've accounted for that in your evaluation when you say the results are under the melting temperature, you've accounted for the degradation of melting temperature for burnup of the UO2 and for the addition of gadolinia, right?

MR. H. KIM: My name is Hung Jin Kim for KEPCO NF.

Gadolinia melting temperature is calculated from the UO2 melting temperature minus the gadolinia rate percent. The UO2 melting temperature is 5080 and then minus the 58 Fahrenheit for 10 megawatt burnup and then minus the gadolinia rate percent.

So, if we if -- so, gadolinia melting temperature for gadolinia can be calculated from the UO2 melting temperature minus gadolinia rate percent.

MR. K. KIM: Thank you.

And, to assure appropriate cooling for APR1400, as its core is maintained greater than the minimum and less than the maximum it is a 100 percent and 115 percent of the design.

And, instability in the core for in the APR1400.

The typical for APR1400 from a hydraulic and nominal conditions are given the table at the lower part of this slide.

The main core of the APR1400 is the coolant system depicted as the blue arrows on the schematic diagram on the right side of the slide enters into the -- with the pressure vessel through the nozzles, pass through and 180 degree turns into the lower core and assembly -- core fuel assembly area and to the plates and the flow out to the bottom nozzle.

Flow pushing on the flow which is not effective for core cooling is up to 3 percent for APR1400, depicted as a red small wave arrows in the diagram.

MEMBER SKILLMAN: How is that 3 percent confirmed?

MR. K. KIM: Just the calculation of a flow network. It's up on the upper level pressure drop information to the core.

MEMBER CORRADINI: So, I was thinking the same question. So, how is it confirmed? So, that's what you calculate. Is there a way to back calculate from measurements upon operation that you're close to what you thought it was?

MR. JANG: My name is Ho Cheol Jang from KEPCO E&C.

We calculate the core analytically any without measurement. It cannot be measured in tests, you only calculate it analytically.

MEMBER MARCH-LEUBA: So, do you have a core exit thermal couples? Core exit temperature?

MR. JANG: Core exit temperature --

MEMBER MARCH-LEUBA: Inside the vessel.

MR. JANG: -- is 16150.

MEMBER MARCH-LEUBA: Right. No, what I'm saying is you can do some calorimetric calculation see what the flow is.

MR. JANG: Yes, yes.

MEMBER MARCH-LEUBA: And, it would be wise to do it occasionally.

MEMBER CORRADINI: Well, but I guess my question -- I'm assuming this fuel design is what's at Shinkori. So, I'm curious, were any measurements made at Shinkori to do a calorimetry measurement to back estimate what the leakage was?

MEMBER MARCH-LEUBA: Yes, but it would depend on what your temperature measurement is.

MEMBER CORRADINI: Sure, sure, but I'm just curious.

CHAIRMAN BALLINGER: Yes, I'm going to expose my ignorance, but, assuming that you could do a calorimetric analysis, you could fix the flow, measure the thermal output and there'll be an imbalance and then adjust the flow to get the right thermal balance. And, that is the sum relationship to bypass flow. Right?

MEMBER CORRADINI: Right. That's what I was thinking.

MEMBER MARCH-LEUBA: And, the active core flow is about controls through DNBR and your limits. So, if this 3 percent is never confirmed, you have an uncertainty on your DNBR.

MR. JANG: Yes, the best estimate of bypass flow rate is 2.4 percent. And, we add the thermal to assure it's not go by this flow is below the original barrier of 3 percent.

MEMBER MARCH-LEUBA: But, your calculation says it's 2.4 percent?

MR. JANG: Yes.

MEMBER MARCH-LEUBA: I claim it's 6 percent and neither you nor I have any basis for it because we haven't measured it.

MR. JANG: You say the 6 percent is your calculation is that?

MEMBER MARCH-LEUBA: Yes, I'm just making the number up. I'm saying, why is it not 6 percent? It begs for a measurement. I mean, that's a very difficult calculation.

MR. KIRCHNER: If it were that large, they wouldn't get the power out the --

MR. JANG: The core limit is different.

MEMBER MARCH-LEUBA: No, the flow is in there. The flow -- the total flow, a 100 percent of the flow goes out at hot layer. And the total power goes out the hot layer. So, as far as the turbine is concerned, you don't know how much it was.

MEMBER KIRCHNER: If it were very large, they would have to increase the core thermal power and you could detect that. But for a very large bypass would require a significantly higher core power output to get the rated power out of the plant.

MEMBER MARCH-LEUBA: No, the same power

goes through the steam generator. All the power goes through the steam generator.

CHAIRMAN BALLINGER: But, TH would go up.

MEMBER MARCH-LEUBA: Your flow would be you'll be burning your fuel.

MEMBER SKILLMAN: That's what I'm saying, you'd have to raise the flow.

MEMBER MARCH-LEUBA: No.

MEMBER SKILLMAN: I asked that question from a background of knowing the importance of the vessel model flow test. The flow rate through that gap at the hot leg is based on the manufacturing tolerance of the idea of the reactor vessel outlet vessel in the OD final machine diameter of the core support structure. One grows into the other at the plant heat.

And, if either of those dimensions is off by more than a fraction of a millimeter, that bypass flow will be different than what you assume. That is a manufacturing issue.

But, I think that the answer is you assume 2.4, you add .6 for 3, but you are really depending on your manufacturing tolerances of the reactor vessel final machining and the internals final machining to assure that the growth closes that gap. Is that accurate?

MR. SISK: This Rob Sisk, Westinghouse.

I appreciate the discussion, the dialogue and would like to discuss it further, but I think we need to have our team get together and get the --

MEMBER SKILLMAN: I think so. And --

MR. SISK: I don't think we're prepared to address this in detail --

MEMBER SKILLMAN: And, I would like --

MR. SISK: -- at this time.

MEMBER SKILLMAN: I would like to offer one more comment on the bypass flow.

That bypass flow may be accurate for four pump operation 100 percent power, four pump operation, 100 percent power.

But, your tech specs allow you six hours, six hours with an off normal condition where you can have less than two pumps per loop.

And, under those circumstances, the tabular data from your prior slide may need adjustment. If you run for three pumps, you will find that you will have flow direction different than four pumps forward.

And, so, I don't believe that's reflected in your analyses. So, I'm curious how you will be communicating this information if you have less than four pumps. You are permitted to run for six hours with less than four pumps forward.

That's your tech spec 3.4.

MR. SISK: Again, appreciate the description, good discussion in general. I think we need to get the team, we have several groups of experts that have to get together and prepare -- talk it through to see how it was addressed. I don't think we're in a position to do that at this time.

MEMBER SKILLMAN: Thank you, Rob.

MR. K. KIM: One thing for the APR1400, initial flow rate less bypass flow.

The design basis limit on DNBR and fuel temperatures are maintained by ACO in the technical specification for the most limiting AOO.

Uncertainties for the DNBR calculations are in input to a period the core and to the motor which power distribution and positive relations and so on. And, TORC model and CHF correlation.

KCE-1 CHF correlation was used with TORC and CETOP codes to calculate the DNBR for normal operation and AOOs. This correlation was developed based on the PLUS7 CHF test data.

ACRS reviews the KCE-1 topical reports on December 14th, 2015, last year.

MEMBER REMPE: So, this --

MEMBER MARCH-LEUBA: Okay, you go ahead.

MEMBER REMPE: A couple of things I have.

First of all, this happens often with the staff and there's usually one ACRS member who points out that you attended a subcommittee meeting on December 14th, not a full ACRS meeting. And, it's very important to distinguish that you only had the ACRS subcommittee review the topical report on December 14th. We will be discussing it tomorrow. And, so, it's good to clarify that.

But, during that discussion, we also, when we reviewed the CHF correlation, we talked about the CETOP-D code, not the CETOP code.

And, as I recall, there was a lot of discussion in the write up on the CHF documentation about that the staff had been concerned about TORC-C and CETOP-D not providing the same results.

And, as I recall, KHNP said we were going to come back and show that it was conservative and that they would give similar results. But, in the documentation for Chapter 4, it just said, hey, CETOP and TORC give the same results, just a simpler model and there was no issue.

So, could you explain what the differences between CETOP-D versus CETOP and why D-TOP is fine?

MR. YOON: I am Taae Young Yoon from KEPCO Nuclear Fuel. Actually, the core in this picture, the CETOP-D core is used to record the CETOP-1 and the CETOP-2. It's just two reduce the calculated time or very toward simplified CETOP code.

MEMBER MARCH-LEUBA: From the review of this KCE correlation, the CHF correlation in the subcommittee, it was our understanding that the staff SER will impose a restriction that KCE cannot be used with any CETOP version. It's not approved for use.

And, therefore, if that restriction on the SER is maintained, which I believe it is so far, as far as I know, we cannot see any CETOP results that use the KCE correlation for our review of Chapter 4 because it's not allowed, it's not approved.

Is that correct? I mean, is that my understanding that the KCE correlation is not approved for use with the CETOP code?

MEMBER CORRADINI: Are you asking the staff or them?

MEMBER MARCH-LEUBA: I'm asking them, they're looking for it.

MEMBER REMPE: Again, I just am puzzled and maybe it is more appropriate for the staff, but I was expecting KHNP to come back with -- again, this is based on the transcript. But, basically, there was a gentleman that had been part of the testing program, and I would almost would have guessed, if my memory's correct, he's from Westinghouse, and he said we're going to be dealing with that later in Chapter 4.

And, so, I, again, okay, so you are the person and did I misunderstand you?

MR. SUNG: Yixing Sung from Westinghouse.

Okay, let me just explain the difference in the code.

MEMBER REMPE: Speak very close to the microphone, because I must be hard of hearing.

MR. SUNG: Again, Yixing Sung from

Westinghouse.

So, let me just explain the code. The code, the base code, the codes are TORC codes.

MEMBER REMPE: Right.

MR. SUNG: And, TORC codes, that's reviewed by the staff was the KCE-1 DNB correlation. And, that's the topical report addressed for.

And, when you look at the CETOP, CETOP is really the simplified code based on the TORC code. Okay, now there are different versions. There's a CETOP-1, CETOP-2, that's actually a program protective system.

And, CETOP-D is the one they use for analysis. All the CETOP codes have the benchmark with the TORC code which is the base code with the correlation license.

But, what the benchmark process are, the calibration processes show all these simplified codes provides a conservative result as compared to the approved version of TORC code.

MEMBER REMPE: So, the KHNP Westinghouse position is that CETOP is just fine to use with the correlation and whatever the staff imposed in the restriction, we'll talk to the staff about.

But, out of curiosity, you mentioned

CETOP-1 and 2, but you didn't mention CETOP-D.

MR. SUNG: I mentioned D as for the analysis.

MEMBER REMPE: Okay.

MR. SUNG: When you do the analysis calculation offline, you use the CETOP-D.

MEMBER REMPE: Okay.

MR. SUNG: But, online, you use CETOP-1 and CETOP-2.

MEMBER REMPE: Okay. That helps me at least understand the differences because I was puzzled. So, thank you.

MEMBER MARCH-LEUBA: But, from the administrative point of view, if the staff's SER restriction that thou shall not use KCE-1 with CETOP will cause a serious problem to this review, wouldn't it?

MR. SUNG: I'm not sure of the SER condition. I would assume the condition would be --

MEMBER MARCH-LEUBA: It's in there.

MR. SUNG: -- the CETOP-1 or CETOP used with KCE-1 has to be benchmarked with a TORC which is the approved by the staff.

MEMBER MARCH-LEUBA: But, it hasn't been approved. As far as December 14, was it, they told us it has not -- it was going to be a limitation on the SER.

So, I mean, administratively, I will need to ask you which of these conclusions that you're telling me are Chapter 4 are based on unapproved code, on the use of a code that has not been approved.

MEMBER KIRCHNER: So, in the thermal design methodology report, and I'll just use the abstract, it makes the statement that the application of the KCE-1 CHF correlation with TORC and CETOP codes is in full compliance with the conditions of the SER on the codes and modeling.

MEMBER MARCH-LEUBA: That's not --

MEMBER KIRCHNER: So, I think we have to ask the staff then to clarify.

MEMBER REMPE: Absolutely.

MEMBER MARCH-LEUBA: We have asked the staff very clearly during the review of the CHF correlation on December 14th, and they told us that it was not provided for CETOP and, therefore, we cannot approve it. But, we'll follow up this, right?

MS. KARAS: This is Becky Karas.

So, I understand you'll ask the staff,

but just because I need to bring them the person who did the topical report review for CHF because they are coming tomorrow with that topic report to the subcommittee. But, we expect to ask that this afternoon. I'll make sure they're available as well.

MEMBER REMPE: Please do.

MR. K. KIM: For appropriate the core summary response, cause and the protection system provide reasonable assurance that the design based are not violate for any normal operating condition and AOOs.

As the analytical method applied to APR1400, flow rate is based on the RPS flow resistance and RCP performance of the APR1400.

The thermal margin analysis were performed by TORC and CETOP codes and since the combination of uncertainty methodology previously opposed by NRC to apply to the thermal analysis.

MEMBER SKILLMAN: Would you describe how you determined the system flow resistance?

MR. JANG: My name is Ho Cheol Jung from KEPCO.

Your question is determining the system flow?

MEMBER SKILLMAN: No, system flow resistance.

MR. JANG: The flow resistance is the estimate from the RSS flow model test for APR1400. We have tested that from the AT flow test and we measured the system from this test and we use the measured data for the determination of APR1400.

MEMBER SKILLMAN: Thank you.

MEMBER CORRADINI: So, just to follow on. So, what was the name of the facility you used to measure? I didn't understand.

MR. K. KIM: You say the test was the flow model test for APR1400?

MEMBER CORRADINI: Yes. I didn't understand, what was the facility? What's the name?

MR. JANG: The test facility, we don't have the test facility name, but at the design page we made a flow test from the model of a system data.

MEMBER CORRADINI: Okay. So, then that was going to lead me to another question which is, since this is a design that's already running in Korea, have you done any measurements either under cold conditions or startup conditions where you actually can verify some of these things?

You have a unique advantage of you

actually have something that runs. So, I'm curious, have you made any measurements so that you can actually verify the flow resistance or at least pieces of the flow resistance?

MEMBER MARCH-LEUBA: And, from the philosophical point of view, you operate this plant with so much mark into burnup that, if you were wrong by a factor of two on the bypass flow, you would never a fuel damage because you have so much margin and you never see the worst AOO.

Just the fact that the Shinkori plant runs perfectly okay, doesn't meant that you -- everything is okay. I mean, it begs for a confirmatory calculation with something because it's an important parameter.

If your flow is off by --

MR. SISK: Again, Rob Sisk here.

I appreciate the discussion. It's an interesting question and we'll need to pursue that with the group as a whole.

MEMBER CORRADINI: I mean I look upon it purely academically. You have a full scale experiment. I'm very curious what it tells you.

MR. SISK: I'll have to go back and see what all's available.

MEMBER CORRADINI: Okay.

MR. SISK: I appreciate the discussion, but again, we're not prepared to do that.

MEMBER REMPE: So, could I beat a dead horse to death and go back to the staff's draft SCE on the topical report on the KCE-1 correlation?

It has that there's an RAI and it's updated response to RAI 37443, question 16, the applicant agreed to delete all references to CETOP-D from the technical report and limit the application of KCE-1 CHF correlation to TORC.

So, when I was reviewing Chapter 4, I was puzzled why suddenly it seemed like it was okay to use CETOP? And, that's where I'm coming from.

And, again, I'm quoting what the staff wrote about the RAI response. But, that was our understanding. And, the reason I'm beating the dead horse to death is that we've got a letter we're going to have to issue tomorrow or discuss tomorrow about the CHF correlation.

MR. OH: This is Andy Oh, KHNP.

Member Rempe, do you expect to answer this question to applicant or us the staff?

MEMBER REMPE: Well, I will be asking the staff, but it would be good, again, to really understand KHNP's position is, yes, it's fine to use CETOP because, again, it could wait until tomorrow, but it would really be good to understand that because I'm seeing -- we're seeing a difference between what is in the topical report versus what's here in Chapter 4.

And, because we have to discuss that this week and write a letter, it'd be good to know what the story is. Okay?

MR. SUNG: This is Yixing Sung, Westinghouse.

Okay, this is no different from the traditional plant analysis. When you look at the base code or baseline for licensing, and this is the TORC code.

But, in the design calculation, from time to time we use the simplified code to do the calculation. But, it's up to the designer to verify the simplified code used in true compliance with the approved version of the code.

In this case, the TORC has to be compared with the approved TORC -- I'm sorry, CETOP has to be compared to the TORC approved version with KCE-1 to demonstrate its use is conservative.

But, it does not change what the staff

approved the position of the limits. It just had to be confirmed in the application.

MEMBER REMPE: And, I guess I was hoping to see that confirmation in Chapter 4 because the staff expressed concerns about the numerical schemes why that they had said use only TORC.

And, I guess, again, we'll talk to the staff, too. But, I guess, that they had a reason for saying don't use anything but TORC and I'm not sure I understand why that reason has gone away.

MEMBER MARCH-LEUBA: Yes, and keep in mind when you're doing that whatever you're doing that often we use or you use CETOP, the fast running code, to run all of the power distributions over the core.

You have the whole depletion and you ran all 1000 of them with CETOP and then confirmed the bad ones with TORC. But, that, if CETOP was not approved, how do you know that that was the bad one?

And, I'm not asking for an answer, but keep that in mind when we're addressing the whole thing because the separation is not a clean as you made it look like.

CHAIRMAN BALLINGER: Continue.

MR. K. KIM: Okay.

There are five total RAI questions in Section 4.4. Among eight, two open items are identified based on the corresponding response to RAI questions.

The response to corresponding questions were submitted in the middle of last year and then most things.

This is the end of our presentation for Section 4.4. Thank you.

MEMBER MARCH-LEUBA: I'm sorry, I guess I wasn't paying attention. That second bullet that the mixed cores DCD, can you talk a little bit about that? How do you handle -- I mean, I know you want 150 percent expect to use only PLUS7 but have you addressed the issue of having more than one fuel element type in the core how that would be calculated?

MR. K. KIM: We usually have -- it depends on the functional approved design activities which nuclear design that they have. And, some have a design and something like that.

But start -- at the starting point of the design, we simulate the difference in geometry between the one type of fuel assembly into the high load fuel assembly by TORC. Okay?

Then there is just some of possible some whatever process solution by calculated by TORC then it can be applied to predict with appropriate correlation for each type of fuel geometry.

Inherently, reflect the effect as result of loading pattern for missed core.

MEMBER MARCH-LEUBA: Okay, so --

MR. K. KIM: We just use only limiting one, we apply to one of a more limiting one than others.

MEMBER MARCH-LEUBA: Okay. So, you have a methodology to handle mixed core calculations?

MR. K. KIM: Yes.

MEMBER MARCH-LEUBA: And, that has been looked at by the staff and -- or you just think it's good?

MR. Y. KIM: This is Yun Ho Kim from KHNP.

Actually, we have a history of developing different type of fuel. We use all, and as we develop PLUS7 and all in the transient time, we need that kind of formulation.

But, for APR1400 design, we only use PLUS 7 fuel type. So, this mixed fuel do not apply to our APR1400.

MEMBER MARCH-LEUBA: So, it wouldn't apply to the DCD review? The DCD is only approved for PLUS7?

MR. Y. KIM: PLUS7 only.

MEMBER MARCH-LEUBA: Okay.

MEMBER SKILLMAN: Jose, that item is open item OI 4.4-1 for mixed cores in the SER on page 4.42 of the SER.

CHAIRMAN BALLINGER: Okay. I think this is a convenient -- this next section is on materials. But, I think this is a good time for a break. So, let's break until -- well, we're actually, by my reckoning, way ahead, so, let's take a break until 15 minutes after 10:00.

(Whereupon, the above-entitled matter went off the record at 9:57 a.m. and resumed at 10:15 a.m.)

CHAIRMAN BALLINGER: Okay. We're back in session. Amongst the many side conversations that happened during the break, we've had discussions related to the CHF --

PARTICIPANT: Your mic?

CHAIRMAN BALLINGER: We have had discussions related to the disposition of the use of the CHF correlation. Maybe that is the best way to put it. And so I think before we get started here, I am not sure who should lead the discussion -- the staff? -- on that issue, and so can we have whoever is ready to go on this discussion?

MR. HAIDER: Yes. This is Syed Haider. I am the technical reviewer of the KCE-1 CHF correlation topical report, and I would like to bring some clarification. I was just called by my branch chief, and even though I am supposed to make the presentation tomorrow, but I think I would like to address this issue today.

The staff had approved the application of the KCE-1 CHF correlation with TORC code contingent upon the DNBR limit of 1.124. So essentially, this was the objective of establishing the baseline number with the application of the TORC code. So if the applicant wanted to modify the 1.124 limit, then they would have to stick to TORC code. But however, for the plant safety analysis, if they could demonstrate that the application of CETOP had provided enough margin, and it was more conservative compared to applying the KCE-1 correlation with TORC code, then this should be acceptable to the staff.

MEMBER REMPE: You go first, since you

went -- well, okay. So could you clarify a little bit more, because perhaps it is in the documentation either submitted by the applicant, or it's in the draft SC, but where did they clearly state that they have used this factor of was it 1.14 you mentioned?

MR. HAIDER: 1.124.

MEMBER REMPE: 124, okay.

MR. HAIDER: Yes.

MEMBER REMPE: Where is it stated? Where can Jose and I find it so we comfortable from --

MR. HAIDER: This is in the --

(Simultaneous speaking.)

MEMBER REMPE: -- I like to see with my eyes.

MR. HAIDER: This is in the SER, in the safety evaluation report --

MEMBER REMPE: Okay.

MR. HAIDER: -- for the topical report.

MEMBER REMPE: Okay. And it says hey, they used this value of 1.124 --

MR. HAIDER: 124, yes.

MEMBER REMPE: -- and therefore it is okay to use CETOP?

MR. HAIDER: No, no, that is not what we

are saying in the SER. In the SER, we are saying that the -- that the use of the KCE-1 correlation is approved with TORC code with the DNBR limit of 1.124. That is what we are documenting in the SER. But if in the DCD part, if they do demonstrate that the application of the correlation with CETOP had additional margin, and it was more conservative, then this should be acceptable to the staff.

MEMBER MARCH-LEUBA: Yes, but let me before -- shall I start? Let me read you what the SER says on page 12, Section 4.6. "The staff include a limitation on Section 5.0 of the safety evaluation to clarify that the use of the KCE-1 correlation with any other subchannel" -- by that it means other than torque -- "would require additional review by the NRC."

MR. HAIDER: Yes. That is the staff understanding that -- that the additional review will be conducted in the DCD part, and that --

MEMBER MARCH-LEUBA: Okay. So --

MR. HAIDER: -- it was conducted.

MEMBER MARCH-LEUBA: -- so the question to the rest of the staff, which is reviewed in Section 4, is -- has this review been performed for CETOP and KCE-1?

MR. LU: The answer is yes. It has been done, and as part of the audit and inspection for Chapter 4, we did check that the multiplier -- addition of multiplier demonstrated the code, when it -- really referencing this KCE-1 correlation, they do have additional margin there. So in terms of safety analysis, they can use other codes referencing this one as long as they have additional margin. That is the part that has been demonstrated, that multiplier, and they did not document any of the DCD because that is really just too much detail, and in the audit, as part of the audit, we checked that multiplier.

MEMBER REMPE: So --

MEMBER MARCH-LEUBA: Okay.

MEMBER REMPE: -- is there a document, again, that the staff has that they can show us that says hey, we did check this multiplier, and it would have been nice to even go further and say so it's okay to use CETOP.

MR. LU: Okay. Yes, we were -- we are going to dig into that audit report. We will find out --

MEMBER REMPE: Okay.

MR. LU: -- with the letter about a year

ago.

MEMBER REMPE: Okay.

MR. LU: Okay.

MEMBER REMPE: Because it didn't come up in the discussions on the CHF --

MR. LU: Yes, that is --

MEMBER REMPE: -- correlation.

MR. LU: -- a nice question because --

MEMBER REMPE: Yes.

MR. LU: -- logically, so you have -- we imposed a limitation in the topical report. The purpose and the intention there is you want to change, 1.124, the number, that really came out from their test data, and then it came from that particular computer code to -- to process those test data. That is where 1.124 came from.

However, if other codes in the safety analysis, that is what the -- you know, Dr. Haider had mentioned that if any other code was used in other application or other part of safety analysis, staff needs to review that. That is true, and that is what we did.

MEMBER REMPE: Okay. Thank you.

PARTICIPANT: Can you state your name one more time for the record?

MR. LU: Oh, Shanlai Lu from staff.

CHAIRMAN BALLINGER: Okay. Shall we pick up where we left off?

MR. SISK: We're done with CHF? I didn't know what we --

CHAIRMAN BALLINGER: I think we are.

MR. SISK: Okay.

MEMBER REMPE: But thank you.

CHAIRMAN BALLINGER: Far be it from me to even think about this.

MR. SISK: Then we will move to 4.5.

MR. J. KIM: Good morning, ladies and gentlemen. My name is Jong Soo Kim in KEPCO-E&C. I will briefly introduce on reactor materials. This section refers to materials used in control element ride mechanism and both the internal and core support.

And this picture shows CEDM assembly and the vessel internals and core support. The main function of the CEDM is to control the reactivity by moving control element assembly. The pressure boundary of the CEDM consists of the -- the motor housing and upper pressure housing, and bent stem, and a housing rod. The main -- main function of the internal and core support is to support the fuel assembly and maintain fuel -- fluid pass in that vessel. The vessel internal and core supports are in accordance with ASME Code NG.

The reactor coolant pressure boundary materials in CEDM: the materials for C -- RCPB in both housing assembly and upper pressure housing assembly are Martensitic stainless steel, Austenitic stainless steels, and nickel-base alloys. They comply with ASME Sections III, II, IX and Reg Guide 1.84.

The reactor coolant contact materials in CEDM are in the material of internal components of the CEDM, that is motor assembly and extension shaft assembly, and they are all corrosion resistance. That means they are Austenitic stainless steels and Martensitic stainless steels, nickel-base alloys, or cobalt alloys.

And weld materials in CEDM, they are also corrosion resistant. They are Austenitic stainless steels and alloy 690 equivalent weld metals.

The materials used in CEDMs are essentially identical to those of the operating plants in U.S. and Korea and show good performances and have been tested for lifetime requirements.

CHAIRMAN BALLINGER: Now, does that -- does that include the use of cobalt-base alloys? When you say essentially identical to Palo Verde and 12 operating plants, I will take your word for it that they also use cobalt-base alloys?

MR. J. KIM: Yes, they do.

CHAIRMAN BALLINGER: It's like 6(b) or something? I think I remember.

MR. J. KIM: Yes, yes, yes.

CHAIRMAN BALLINGER: And that experience has been okay, and it hasn't resulted in issues related to radioactivity in the coolant and things like that?

MR. J. KIM: Yes. They included cobalt-based alloys. So far, we don't have any material problems in the CEDMs in Korea, no.

CHAIRMAN BALLINGER: Okay. Thank you.

MR. J. KIM: And various provisions of materials and verification processes are applied on Austenitic stainless steels of CEDMs, and basically, requirements for RCPB materials of DCD Section 5.2.3.4 are also applied for CEDM Austenitic stainless steels.

And venting of CEDM will be applied for degassing before the plant starts.

And other materials of the nickel-base alloys, cobalt alloys, and Martensitic stainless steels are used for intended special purposes for springs, grippers, latches and links, and inserts and tabs and balls. And those materials have been used with satisfactory performance in plant operations.

Reactor internals and core support materials: internals and core support materials comply with ASME Code Section NG and Reg Guide 1.84. They are primarily Type 304 Austenitic stainless steels. Fasteners are typically Type 316 Austenitic stainless steels, and hardfacing wear areas and controls on cold-worked Austenitic stainless steels are also applied. The material used in reactor internals and core support are proven materials and showed good performance in plant experience. And material specifications, for core support barrels, upper guide assembly, core shroud assembly are mainly Austenitic stainless steels.

MEMBER RICCARDELLA: Excuse me. The baffle-former assembly is welded, I believe? You don't have baffle-former bolts?

MR. J. KIM: Yes, welded.

MEMBER RICCARDELLA: Thank you.

MR. J. KIM: Bolt -- bolt and pin materials are Austenitic stainless steels. Chrome plating and cobalt hardfacing materials are also applied, and type 308L Austenitic stainless steel reactor material is also used for welding. And the welding and nondestructive examination comply with ASME Section III, NG-4000, and NG-5000, respectively.

The requirements for the RCPB materials of DCD Section 5.2.3.4 and Reg Guide 1.44 are applied on Austenitic stainless steels of reactor internals and core supports in order to avoid sensitization, and other materials of Austenitic stainless steels and Martensitic stainless steels are used for intended purpose of alignment keys, pins, and HJTC tube and hold-down rings, and these other materials showed good satisfactory performance in plant experiences.

The challenging degradation for reactor internals and core support materials are irradiation-assisted stress corrosion cracking and void swelling. The assessment -- assessment for APR 1400 reactor internals and core support materials are performed -- was performed using EPRI methodologies, and the result was acceptable.

Open items: there were 23 RAIs for this section. Two RAIs are -- remain as open items. They are about Versa Vent system of CEDM. KHNP was questioned that the Versa Vent should be regarded as a pressure boundary item. This is number 15. And it was also requested to provide data or operational experience that demonstrates the Versa Vent can work in -- in practice -- in practice to eliminate the air of the CEDM. This was 16. And the responses of these open items were submitted.

This is --

CHAIRMAN BALLINGER: I have a question about the Versa Vent. Am I to understand that these Versa Vents are also used for example at Palo Verde and other places?

MR. J. KIM: It was used, but for now, the Palo Verde does not use any more when they change it. They -- or the Versa had.

CHAIRMAN BALLINGER: Okay. Thank you.

MR. J. KIM: And this -- yes. This is all for Section 4.5. Thank you for your attention.

MR. SCHULTZ: Thank you. Before we go to the next section, leave reactor materials, could you go back one slide? Just a general question I have: as you have applied the EPRI-developed software for the assessment of the reactor vessel internals, what does that mean? In the application, or the application of that software, is it a best estimate evaluation, or are there conservatisms in the input data that one uses to provide assurance that the performance is going to be satisfactory? How is that evaluation done? Can you provide some additional detail on that?

MR. J. KIM: Yes. The additional detail was in the report.

MR. SCHULTZ: Yes.

MR. J. KIM: Yes yes --

MR. SCHULTZ: I haven't seen --

MR. J. KIM: -- yes yes --

MR. SCHULTZ: -- the report yet --

MR. J. KIM: -- yes --

MR. SCHULTZ: -- but --

MR. J. KIM: -- and we -- we referenced the methodology and the EPRI-developed software used in that, yes, used in that, that was developed by the EPRI and -- EPRI.

MR. SCHULTZ: Yes. And do you demonstrate that the assessment results are acceptable by doing what you would call a bounding analysis or conservative, or do you do a best estimate evaluation of the performance over a certain period of time and then say, oh, it came out okay?

MR. J. KIM: Yes.

MR. SCHULTZ: Can you --

MR. J. KIM: Yes.

MR. SCHULTZ: -- elaborate there?

MR. J. KIM: -- actually, there was the recommended acceptance criteria in the EPRI guidelines there.

MR. SCHULTZ: The acceptance criteria on the --

MR. J. KIM: Yes --

MR. SCHULTZ: -- result?

MR. J. KIM: -- yes, yes, that was --

MR. SCHULTZ: And does it specify how to determine how one does the evaluation in terms of input parameters?

MR. J. KIM: Yes. For example the -- the -- irradiation-assisted stress corrosion cracking, there -- they provide the susceptible stress.

MR. SCHULTZ: Okay.

MR. J. KIM: Yes, and they provide the ratio, yes, provide some reference stress, and --

and if you are below those stress, and we -- we are -- think it is acceptable. And -- and for the -- the void swelling, yes, they also provide some acceptance criteria, volumetric 2.5 percent, 2.5 --

MR. SCHULTZ: Do you --

MR. J. KIM: -- yes.

MR. SCHULTZ: Yes. Do you compare your results here with what has been done for other CE-type plants that -- is there some sort of database or results evaluation that has been done for other similar plants --

MR. J. KIM: I --

MR. SCHULTZ: -- that is used for comparisons, or is it just one evaluation for the APR1400 that you have done --

MR. J. KIM: Yes, actually, we did just follow one evaluation for the APR1400. This is first time in Korea for --

MR. SCHULTZ: First time for you to --

MR. J. KIM: Yeah yeah yeah --

MR. SCHULTZ: -- use this?

MR. J. KIM: -- yeah, for this kind of analysis.

MR. SCHULTZ: Yes. Did you have any support from EPRI in reviewing the results to get some assurance that the results that you have developed is typical --

MR. J. KIM: We --

MR. SCHULTZ: -- of what they, the developers of the tool, would have expected?

MR. J. KIM: Yes. We also contact with EPRI because we need the -- the software.

MR. SCHULTZ: Yes.

MR. J. KIM: Yes, so that we contact with EPRI people, yes, with -- with this work. But they just provide the -- it is for free, yes, they provide for anyone because we are EPRI members.

MR. SCHULTZ: Yes.

MR. J. KIM: Yes.

MR. SCHULTZ: But did you have any EPRI staff or consultants review what you had done with the tool?

MR. J. KIM: With -- with the -- with -- to -- we did not consult with the result, yeah. But that they provide the detailed methodology and those -- and their detailed procedures, yes.

MR. SCHULTZ: Okay. Okay.

CHAIRMAN BALLINGER: But --

MR. SCHULTZ: That is --

CHAIRMAN BALLINGER: -- you --

MR. SCHULTZ: -- helpful.

CHAIRMAN BALLINGER: -- also had access to all of -- because you're EPRI members, you have access to all of the -- the MRP, the Materials Reliability Program data and all the reports and everything to -- to augment what -- what they are -- what you are doing, is that correct?

MR. J. KIM: Yes. We can -- we did contact with EPRI when we need some help or documents or -- or we -- if we have some when we feel some need for discussion, yes, yes we did.

MR. SCHULTZ: So in summary, you are satisfied with the results that you achieved from applying this methodology because, as you say in the first bullet --

MR. J. KIM: Yes.

MR. SCHULTZ: -- this is a very challenging area --

MR. J. KIM: Yes.

MR. SCHULTZ: -- and it is something that you want to be able to demonstrate that you have in terms of your material selection and the parameters that are influencing factors in this --

MR. J. KIM: Yes.

MR. SCHULTZ: -- in this area --

MR. J. KIM: Yes.

MR. SCHULTZ: -- that you have got it covered.

MR. J. KIM: Okay.

MR. SCHULTZ: You are not going to experience failure due to the mechanism?

MR. J. KIM: Yes.

MR. SCHULTZ: Good. Oh, I will look forward to reviewing the report. Thank you.

MR. J. KIM: Okay, thank you.

CHAIRMAN BALLINGER: In addition, are these materials similar to materials that have been used in the OPR1000?

MR. J. KIM: For the analysis?

CHAIRMAN BALLINGER: No, just in practice.

MR. J. KIM: Materials, yes, yes, yeah, yeah, identical, yes. Some -- I mean, there are some weld materials, for example, alloy 600 to the 690 --

CHAIRMAN BALLINGER: Wise move.

MR. J. KIM: Yeah, yeah, yeah, yeah, yeah.

CHAIRMAN BALLINGER: Very wise move.

MR. J. KIM: Yeah.

CHAIRMAN BALLINGER: Okay. Okay. So but the experience has been good in OPR1000 for materials that you are now using in APR1400?

MR. J. KIM: Yes, yes, that is correct.

CHAIRMAN BALLINGER: Because there is a long history there.

MR. J. KIM: Yes.

CHAIRMAN BALLINGER: Okay. Thank --

MEMBER SKILLMAN: Jong --

CHAIRMAN BALLINGER: -- you.

MEMBER SKILLMAN: -- Jong Soo, let me ask this --

MR. J. KIM: Yes.

MEMBER SKILLMAN: -- question: Dr. Riccardella asked whether or not the baffles are welded or bolted, and you responded that they are welded.

MR. J. KIM: Yes.

MEMBER SKILLMAN: So there are no baffle bolts.

MR. J. KIM: No.

MEMBER SKILLMAN: Okay. What experience is there to show that the welded design is a durable design?

MR. J. KIM: In the -- actually, the --

the design of reactor internal is in another different section, 3.9.5. That is a more -- there is more detailed information about the reactor vessel internals. And so -- so far, we -- the reactor vessel internals, we just don't have any material issues of APR1000, so we believe this is proven materials for the reactor vessel internals.

MEMBER SKILLMAN: Are the APR1000 internals all welded?

PARTICIPANT: Yes.

MR. J. KIM: Yes, yes.

MEMBER SKILLMAN: No bolts?

MR. J. KIM: No bolt. It is -- it is CE type, no bolt.

MEMBER RICCARDELLA: Dick, you know, the welded design is similar to the CE plants in the U.S., and there have been no incidences of issues with the welded baffle form of design in CE plants, with the exception of Palisades, which happens to have bolts.

MEMBER SKILLMAN: Okay. Thank you.

MR. J. KIM: Chapter 4.6 is for functional design of the reactivity control system of APR1400. The section 4.6 describes the control rod drive system, the CRDS. The CRDS consists of control element drive mechanism and digital rod control systems. The CEDM is for inserting or withdrawing the CEAs, and the DRCS is for actuating the CEDMs.

The information and evaluation of combined -- combined performance of the reactivity control system are about design basis events which are analyzed in Chapter 15. Chapter 15 requires reactivity control system to operate for preventing or mitigating each event. And there are no open items for Section 4.6. Yes? Yes?

MEMBER SKILLMAN: Question, please. In the material, in our reading material, the full-strength rods have a tip, and that tip is a B4C pellet wrapped in a sleeve of 347 material, and it is called felt material.

MR. J. KIM: I am sorry. I don't understand.

MEMBER SKILLMAN: See, the text that I will read describes my question. "The applicant stated that the control elements of a four-element or twelve-element full strength CEA consists of an Inconel 625 tube" --

MR. J. KIM: Yes.

MEMBER SKILLMAN: -- "loaded with a

stack of cylindrical boron carbide pellets, B4C, with the exception of the lower portions of the elements, which contain reduced diameter B4C pellets wrapped in a sleeve of Type 347 stainless steel."

MR. J. KIM: Yes.

MEMBER SKILLMAN: What is your experience with this design?

MR. J. KIM: Yes. For the reactor internal and cesium -- yes, I am sorry, yes, there is some -- another --

MR. YOON: Tae Young Yoon from KEPCO NF. There is some material CEA, right? The CEA design is same as the System 80+ design. It -- it has been used a lot of years, and we also, CEA design you see in OPR1000 in Korean plant also, yes. It has a lot of experience for that.

MEMBER SKILLMAN: Thank you. Thank you. Okay.

MEMBER MARCH-LEUBA: Off the top of your head, or anybody in the room, do you remember what the worth of the highest rod is, the highest worth of a rod is? I am thinking rod injection: did you know how much they are worth?

(Pause.)

MR. CHON: This is Woochong Chon from

KEPCO Nuclear Fuel. It is around 4,000 PCM.

MEMBER MARCH-LEUBA: Okay. Very little, yes, okay. Thank you.

CHAIRMAN BALLINGER: Okay. Continue, okay.

MR. J. KIM: Summary: the APR1400 reactor design full demonstrates to comply with requirements of federal regulations and NRC regulatory documents. There are no open items for Sections 4.1, 4.3, and 4.6. There are six open items in total for Sections 4.2, 4.4, and 4.5. Thank you. Thanks for your attention.

CHAIRMAN BALLINGER: We're mercifully ahead of schedule, and consulting with my staff, okay. We are ahead of schedule, and we've discussed it with the staff, and they are ready to pick up and start so that we can get -- stay ahead of the game, if you will. So thank you very much. Thank you very much.

MS. KARAS: This is Becky Karas. We are missing I think --

CHAIRMAN BALLINGER: Oh, oh --

MS. KARAS: -- one staff member, so we're trying to get a hold of them, but we can get through I think the first -- we can get through 4.2.

When we get to 4.3, we are missing one staff member that we have been trying to get a hold of now to come back early.

CHAIRMAN BALLINGER: Okay. Thank you, distance made good.

MR. WUNDER: Thank you, Mr. Chairman, and good morning, ladies and gentlemen. I'm George Wunder, the project manager assigned to Chapter 4 for the APR1400 review. I'm joined today by Chris Van Wert and Alexandra Burja of the Reactor Systems Branch, and by Andrew Bielen of the Reactor Systems Analysis Branch in the Office of Research.

When we switch out, we're a little bit constrained by space today, but when we switch out, I'll be joined by Jim Gilmer and Carl Thurston, also of the Reactor Systems Branch, and finally by Jonathan Honcharik and Dan Widrevitz both of the Mechanical and Chemical Engineering Branch.

CHAIRMAN BALLINGER: Excuse me, I violated the prime directive. I'm told that we need to tell the contractor to mute his or her phone.

MR. WUNDER: The review was conducted by a team of eight members of the technical staff. They are the presenters who I have just introduced to you, and also Peter Yarsky of the Reactor Systems

Analysis Branch Office of Research.

We'll be presenting to you today in the areas of fuel system design, nuclear design, thermal and hydraulic design, materials, and reactivity control. I'd now like to turn the presentation over to Chris Van Wert. Chris?

MR. VAN WELT: Good morning, everyone. My name is Chris Van Wert and I will be presenting the staff's review of - sorry. All right, good morning, again. My name is Chris Van Wert and I will be presenting the staff's review of DCD Section 4.2 fuel system design.

I've listed on this slide the areas of review, and this is similar to the other sections that you'll hear about today, but we covered design bases, descriptions and design drawings, design evaluation, the testing, ITAACs, and COL action items and certification restrictions.

And now I'd like to discuss a couple of the more challenging review areas that we encountered during the review. The first one is burnup dependent thermal conductivity degradation, and I believe that was discussed a little bit this morning.

And as you are well aware, the APR1400's fuel system design is based on the use primarily of the FATES-3B fuel design code which does not contain a burnup dependent TCD model. The staff identified this as a concern regarding compliance with GDC 10 for various fuel system damage and fuel rod damage mechanisms, as well as 10 CFR 50.46 for core coolability requirements.

It should be noted that the burnup dependent TCD model review is addressed more completely within the associated topical report reviews for the PLUS7 fuel design as well as the large-break LOCA topical report. The resolution of the DCD Section 4.2 open item is therefore dependent upon the successful completion of those reviews. Any questions on that? Yes?

MEMBER REMPE: Yes, could you just talk a little bit about what the concerns are and, I mean, typically when - are they going - my understanding from the documentation is they're not going to change their code and they went and used FRAPCON and did something, but they concluded that things are okay, and is the staff saying, "No, we want a penalty," or what's going on and what are the issues?

MR. VAN WELT: Yes, yes, so this is

ongoing, so I don't want to give the impression that it is complete and set in stone at this point, but the path forward that we are looking at right now is a penalty which would be applied to the results from FATES-3B. The penalty is based on comparison of FATES-3B results to the Halden test data, as well as the staff performed confirmatory runs using FRAPCON.

So they are not relying on FRAPCON as a design basis code. That is the staff's confirmatory tool while reviewing their - I'll say the new methodology that they're proposing is FATES-3B plus penalty, and so we're using FRAPCON just to confirm that.

MEMBER REMPE: Okay, thank you.

MEMBER MARCH-LEUBA: The penalty is on the temperature or on decay? Can you explain a little more?

MR. VAN WELT: I would like to hold off on the specifics of it just because that part is still ongoing.

MEMBER MARCH-LEUBA: Okay.

MR. VAN WELT: And that is associated with the PLUS7 topical report.

MEMBER MARCH-LEUBA: We'll just call this a general described penalty.

MR. VAN WELT: Yeah, but to give you a little nugget, there is a temperature penalty, but we won't go into that on the other topical reports as they come up.

MR. SCHULTZ: Okay, I'd just like to repeat back so I fully understand. The applicant is going to derive and propose a penalty?

MR. VAN WELT: Correct.

MR. SCHULTZ: And the approach that they've taken and present to you for that penalty and its application will be for them to do. The confirmatory analyses that are being done with FRAPCON, have they been completed so you have a sense as to what those results are in comparison to the FATES code?

MR. VAN WELT: Yes, yes, we are actually well along in this process. We've been having probably every other week public phone calls with KHNP on this topic, and there have been different penalties proposed as we've gone along, and we've performed confirmatory runs on those, and so I might have made it sound like we were done.

MR. SCHULTZ: No, you didn't.

MR. VAN WELT: Okay.

MR. SCHULTZ: My next question is what's

the schedule here because it certainly involves a submittal as well as a review?

MR. VAN WELT: Correct.

MR. SCHULTZ: I mean, you've got some of the calculations done, but -

MR. VAN WELT: Yes.

MR. SCHULTZ: - the most important thing is to get a thorough review of what's submitted.

MR. VAN WELT: So we have performed all of the necessary FRAPCON confirmatory runs. Right now - and I believe there's pretty good agreement between KHNP and the staff on the temperature side of the equation.

Right now, I believe KHNP is looking at the impacts on the large-break LOCA PCT calculations, and we have some internal trace calculations that we will use for confirmatory comparisons when that comes in.

I will casually glance over at KHNP at this point to see if everything is on schedule for the large-break LOCA PCT calculations, but I believe we're looking in a few weeks, sorry, three months. I was just signaled.

MR. SCHULTZ: Okay.

MR. VAN WELT: But that would be the

final results.

MR. SCHULTZ: All right.

MR. VAN WELT: So we will probably in a number of weeks, a month or two, we should have the preliminary results that we can use to compare against.

MR. SCHULTZ: Has the staff reviewed the data set, the Halden data set? It's just going to be Halden data that will be used?

MR. VAN WELT: For the temperature, yes.

MR. SCHULTZ: The applicant doesn't have any separate data that could be used?

MR. VAN WELT: Correct, there's no bilateral data that was supplied.

MR. SCHULTZ: Okay, and has the staff reviewed the extent of the Halden data that's going to be applied -

MR. VAN WELT: Yes.

MR. SCHULTZ: - and agreed that that data set is sufficient?

MR. VAN WELT: Correct, yes.

MR. SCHULTZ: Thank you. And that information has been submitted or is it going to be submitted all in one package?

MR. VAN WELT: It will be submitted in

one package.

MR. SCHULTZ: Okay, in about three months?

MR. VAN WELT: I will look over, but I believe that is -

MR. SCHULTZ: That's close enough.

MR. VAN WELT: Yes.

MR. SCHULTZ: Thank you.

MEMBER REMPE: So I'm a little slow because I was going to pull up my schedule, but isn't Chapter 15 coming to us? Will it come to us before you have this completed?

MR. VAN WELT: I believe it will be very similar to what you're seeing today where the DCD Chapter 15 will be like just DCD Chapter 4 where we have an open item associated with the TCD issue, and so unfortunately, that means you will also have maybe a -

MEMBER REMPE: Okay.

MR. VAN WELT: - bigger gap than you would like at that stage, but during the topical report, you know.

MEMBER MARCH-LEUBA: On the plan, I believe, it's to adjust the acceptance criteria by putting a penalty in temperature, so you don't have to redo the calculations. Is that the way you're thinking?

MR. VAN WELT: For? Now is this -

MEMBER MARCH-LEUBA: For the LOCA PCT calculations.

MR. VAN WELT: It's actually redoing the LOCA PCT calculations. That's the three months that we're talking about.

MEMBER MARCH-LEUBA: Oh, for LOCA they're not going to use a penalty. They're going to do the calculations.

MR. VAN WELT: There's a penalty on the calculations, but they are re-running because the - I don't know if -

MR. CHON: This is Woochong Chon from KEPCO Nuclear Fuel. We will add the fuel PCT penalty on the fuel, and that data will be transferred to the calculation of the RELAP for the large-break LOCA calculation, so we have to re-perform the whole large-break LOCA calculation with the added penalty fuel data.

MEMBER MARCH-LEUBA: Yes, it's not just an acceptance criteria. You have the thermal hydraulics, so you have to follow-up with a penalty.

MR. CHON: We have to perform whole

calculation.

MEMBER MARCH-LEUBA: Yeah, so, well, to perform the calculations, you need to know what the penalty is.

MR. VAN WELT: Yes, and we've come to an agreement on that. That's the part where we're talking about the FATES-3B penalty, but that all feeds into the input decks that they need, and that's the part they're re-running at this point.

MEMBER MARCH-LEUBA: Talk about it whenever Chapter 15 comes.

MEMBER REMPE: That is coming in March subcommittee week, and will the RELAP analysis be completed or we're going to be doing Chapter 15 without the RELAP analysis results?

MR. CHON: This is Woochong Chon of KEPCO Nuclear Fuel again. To finish all of the large-break LOCA calculations, we expect it will take three months, but before that, we can check the sensitivity study with the penalty added fuel temperature, so we can calculate earlier than three months, but the final DCD revision will be done after three months later after the results of the TCD.

MR. SCHULTZ: Why would it only effect

large-break LOCA calculations?

MR. VAN WELT: It doesn't. There are other impacts.

MR. SCHULTZ: Okay, those will be identified as well?

MR. VAN WELT: Yes, in general, some of the other designs. The DCD Section 4.2 is really more of the high-level results -

MR. SCHULTZ: Understood.

MR. VAN WELT: - calculation, so the topical report will -

MR. SCHULTZ: I was thinking of Chapter 15.

MR. VAN WELT: Oh.

MR. SCHULTZ: And how many pieces are involved there and affected there. Thank you.

MR. VAN WELT: Yes. If there are no open questions on TCD, we can -

MEMBER KIRCHNER: It begs a question that's a bit of a digression, but how has this been handled with the CE plants? I would assume they're using CE derivative methodologies and so on.

MR. VAN WELT: Correct.

MEMBER KIRCHNER: The FATES code. This is a thermal degradation. The gap conductivity has been known for a long, long time, so what was done previously with the US CE plants to address this?

MR. VAN WELT: Well, keeping in mind that the staff only became aware of the magnitude of some of the deltas more recently, especially when you compare it to the licensing basis for most of the CE fleet, so must they have command for an LAR since the staff became aware of this, they would not have necessarily had to address it in the same magnitude that KHNP is addressing here.

That being said, you could see certain plants have come in with LARs and they have addressed it, but, and I don't want to speak for NRR on this topic. You know, I would defer to them on it, but my understanding was that there was -

Well, actually, Westinghouse might also be able to speak up on what manner the generic communication went out, but there was, I think it was tied to 50.46, but it was the impacts of TCD on that, and the operating fleet did have to come in and address more or less a justification for continued operation, a rough approximation of the impact, and I believe most of them have committed to switching over to the newer codes that are coming out or have come out which address this.

So for Westinghouse, there's PAD5. For AREVA, there's COPERNIC. For GE, it's PRIME. As those new codes have been coming out, the operating fleet has been switching the licensing basis over to the newer codes.

MEMBER MARCH-LEUBA: And I realize that you're not the right person to ask this, but I would expect that didn't they evaluate it through Part 21 and suddenly your calculation of record is put in question? So I'm sure if they already planned the Part 21 evaluation if, "Does this affect me?"

MR. VAN WELT: And there might be the Part 21 evaluations for that, especially when you get down to what different plants did or the plants as a whole did. I would want to refer to NRR for that one.

MEMBER MARCH-LEUBA: No, no, because we are reviewing Part 5 soon.

MR. VAN WELT: Yes, yes, I think the draft SDR is already completed, so.

MEMBER MARCH-LEUBA: Yes, we will see them in May.

MR. VAN WELT: Okay, excellent. Any more TCD questions? Okay, we can go to the - oh, excellent, so the next challenging area - was there

- okay, the next challenging review area that I would like to discuss is the fuel assembly structural response to externally applied loads.

And I believe you heard about this this morning as well, but during the review of the structural response analysis, the staff noted that the reference methodology was not strictly followed in its entirety, and that caused us to question the determination or load limits for the PLUS7 fuel assembly.

As a result of that concern and the back and forth between the staff and the applicant, the applicant is now in the process of completing its open item resolution plan which includes a comprehensive test program of the PLUS7 fuel assembly and its modeling of both the beginning of life and end of life, simulated end of life conditions.

Again, this is another ongoing area, but the staff has been auditing the tests as they've occurred, and I believe the next test that we are going to witness is coming up shortly at the end of February here, which will be related to the flow dampening credit.

And I believe the current schedule is

the end of spring, maybe early summer, we should have the completed documentation for this and we should be able to complete a review at that time. Yes?

MEMBER SKILLMAN: What does test mean in that context?

MR. VAN WELT: So they are doing actually a full suite of tests including grid assembly tests. They were doing through grid, BOL, and EOL conditions, SEOL conditions, simulated EOL conditions where they relaxed the springs to simulate the grid relaxation that would occur.

Right now, the test that I mentioned that's coming up is using the Westinghouse VIPER loop which normally is used for flow induced vibration test. They're modifying that, or did modify it to perform flow dampening analysis or testing so they can take credit for that in the revised analysis when they present that.

MEMBER SKILLMAN: So these are physical tests from which data is extracted for evaluation for confirmation?

MR. VAN WELT: Correct.

MEMBER SKILLMAN: Copy that. Thank you.

MR. VAN WELT: Thank you. So when I was

looking at this, I realized it sounds a little bit odd here. So the title of this slide is, "Fuel system design meets the following requirements," and the next sentence says, "The staff is currently unable to make regulatory findings on the fuel system design."

The two open items that we talked about are rather large, and since they impact pretty much all of GDC 10, 35, GDC 2, we're unable at this time to make regulatory findings. However, we do have a plan. The applicant is well on the path to reaching the completion there, and we expect to be able to make our regulatory findings very soon.

Any further questions on fuel system design? Thank you very much.

MS. BURJA: All right, my name is Alex Burja and I will present to you the staff's review of DCD Section 4.3 nuclear design. First, I will go over the areas that the staff reviewed, then I'll go into a little bit more detail on some of the more challenging areas. Finally, I will provide the staff's conclusions before turning it over to Dr. Andrew Bielen from research who conducted the staff's confirmatory analyses.

The first area that the staff reviewed

under the nuclear design was the design bases to ensure that they are consistent with the general design criteria, which indeed they were.

Under the area of power distributions, the staff looked at the representative power distribution calculations presented in the DCD to ensure that they were comprehensive and that they included axial, radial, and pin power distributions, and accounted for different times in the cycle, as well as different operating conditions.

The staff confirms that these power distributions that were presented were well within the power distribution design limits, and based on the confirmatory analyses, the power distributions are about what we would expect.

The staff also ensured that there are means to monitor the power distributions. These included the non-safety related core operating limits supervisory system or COLSS, which notifies operators if tech specs are exceeded, and the safety related core protection calculator system or CPCS which will initiate a reactor trip if subpoints are exceeded.

The staff also ensured that there are means to control the power distributions, and these consist of the core loading pattern, control element assemblies or CEAs, and soluble boron.

The next major area of review for the staff was that of reactivity coefficients.

MEMBER MARCH-LEUBA: Alex?

MS. BURJA: Yes?

MEMBER MARCH-LEUBA: On the power distributions, my understanding for APR1400, what they do is they run a full 18-month depletion of the core and calculate every single power distribution every day of those 18 months, and then they use those power distributions to calculate the limits. Is that correct?

MS. BURJA: That's my understanding, yes.

MEMBER MARCH-LEUBA: Yes, so that's what you mean when you reveal they were presenting power distribution. You saw every power distribution in an 18-month cycle, not that you looked at every picture, but they were there.

MS. BURJA: Right, right, everything that was presented in the DCD.

MEMBER MARCH-LEUBA: Okay, thank you.

MS. BURJA: Okay, so in the area of reactivity coefficients, the staff ensured that there is a negative power coefficient of reactivity including a negative moderator temperature coefficient at power, and the staff also ensured that the transient and accident analyses used conservative values relative to what was presented in Section 4.3.

MR. SCHULTZ: So this statement doesn't apply. We shouldn't presume that you've got - you performed the conservative value evaluation that you would have for Chapter 15 events, or is this blanketing that conservative values have been used or will be used in Chapter 15 analyses?

MS. BURJA: So based on the - so Chapter 15 tells us what MTC and FTC were used, and I compared those values to what's used in 4.3 -

MR. SCHULTZ: Okay.

MS. BURJA: - and made sure that the values used in the accident analyses are conservative relative to the design values.

MR. SCHULTZ: I see, thank you.

MS. BURJA: Does that make sense?

MR. SCHULTZ: That's fine.

MEMBER MARCH-LEUBA: I know we are not reviewing Chapter 15 now, but do they not use a 3D automatic calculation of those coefficients for the accident analysis or do they use the point kinetics with the feedback with the conservative feedback?

MS. BURJA: It's my belief that they use the point kinetics.

MEMBER MARCH-LEUBA: Point kinetics?

MS. BURJA: Yes.

MEMBER MARCH-LEUBA: So then it's important that you review that they're using a conservative number.

MS. BURJA: Right.

MEMBER MARCH-LEUBA: One of the being positive for a week or two of every cycle.

MS. BURJA: Right, I did go back and confirm that for accidents like rod ejection, they do use a positive MTC value at the low power conditions.

MEMBER MARCH-LEUBA: Yes, and while we're talking about cross sections, most of the - you are aware of the use in there before cross sections for gadolinia which uses - but they use gadolinia in special treatments, is that correct? Because before, it had element gadolinia which does not deplete properly.

MS. BURJA: So I actually did a little bit of research on this, and based on the Brookhaven

National Lab data tapes, they do have some isotopic data for gadolinia.

MEMBER MARCH-LEUBA: So it was for amplified with the two main gadolinia isotopes, or three, or however. Two are the important ones.

MS. BURJA: Right.

MEMBER MARCH-LEUBA: And you verified that the limit is correct and they depleted the gadolinia correctly because that affects that MTC significantly.

MS. BURJA: Yes, yes, we did look into the gadolinia treatment.

MEMBER MARCH-LEUBA: Okay.

MS. BURJA: Under the area of reactivity control provisions and requirements, the staff looked at things having to do with CEA patterns, insertion limits, worths, and soluble boron and burnable absorber worths and well.

And the staff confirms that the available CEA worth is sufficient to safely shut down the reactor during normal operation and accident conditions, that conservative worth values relative to what was calculated for the 4.3 nuclear design are used in the Chapter 15 transient and accident analyses.

The staff also ensured that the requirements for reactivity controls such as the necessary worths for different times in cycle are clearly defined, and finally, the staff reviewed that there are appropriate limits on reactivity insertion rates to preclude power excursions.

MEMBER MARCH-LEUBA: Speaking of this area, KHNP makes a claim that they are stable with respect to xenon oscillations, that stable oscillations will dampen. Did you find that?

MS. BURJA: Yes, the decay ratio was negative, so they do dampen.

MR. SCHULTZ: Alex, how do you establish that the conservative worth values that are used in the transient and accident analyses are satisfactory? Is there some criteria that the staff has to say they will be called conservative if they are so much larger than nominal, or is it just oh, it's larger than nominal, so I guess that's conservative, or lower than nominal depending on the -

MS. BURJA: So there's no hard and fast rule, but ultimately, you know, as long as they're using something that, you know, if they assume that they need a certain worth and - so if the nuclear design says, "All right, we have this certain worth," they must have at least that much for the transient or - I'm sorry.

I'm not answering this very well, but it is subjective, and based on the design information, the assumptions in the transient and accident analyses have to at least be the same or in a more conservative direction.

MR. SCHULTZ: And then you're going to do the evaluation also compared margin to limits for the final evaluation and analysis, correct?

MS. BURJA: Mm-hmm.

MR. SCHULTZ: Okay, thank you.

MS. BURJA: Are there any more questions on the reactivity control? Okay, the staff also reviewed provisions to prevent reactor criticality during refueling, and these are sufficient such as a soluble boron concentration, enough to hold the core K-effective under 0.95, as well as procedures that require refueling to stop if neutron count rates were to suddenly jump. Next slide, please.

The next area of review was stability against xenon-induced power distribution oscillations, and based on the staff's review, we found that only axial xenon oscillations were possible, and they can be suppressed using the CEAs.

The staff also reviewed the applicant's analytical methods for the nuclear design, and these consist of the DIT and ROCS codes which were first approved by the NRC in 1983, then again in 1988 to include refinements for gadolinia fuel. Because these were previously approved codes, the staff really focused its review of this area on applicability of the codes to APR1400.

The staff also ensured that the codes used to process information from the ex-core detectors for use in the CPCS are appropriate for such an application.

MR. SCHULTZ: The review of the codes in 1988, I presume that included the range of gadolinia that they -

MS. BURJA: Yes.

MR. SCHULTZ: - that they're using.

MS. BURJA: I don't recall offhand, but I know the range meets -

MR. SCHULTZ: Encompassed the -

MS. BURJA: Yes.

MR. SCHULTZ: - eight percent? Thank you.

MS. BURJA: Yes. In the area of reactor

vessel fluence, the staff reviewed the methodology that the applicant used which is DORT, a 2-D discrete ordinates transport code, and also the assumptions used within it to ensure that it's conservative.

The staff also reviewed the vessel fluence calculation and the calculation of bias and uncertainty to ensure that everything was calculated accurately and applied appropriately. And finally, the staff requested a combined license information item for plant specific surveillance data to benchmark the methodology.

The staff also reviewed the nuclear design tests in the initial test program and found that they conformed to Reg Guide 1.68 and are adequate to verify the nuclear design.

And I'd like to point out that there was an open item in the SDR related to a Chapter 14 question, but it's not characterized as a confirmatory item because of the satisfactory REI response, and that ultimately is going to be handled in Chapter 14. And finally, the staff reviewed the nuclear design related tech specs and safety limits to ensure that they are adequate.

MR. SCHULTZ: Alex, the COL information

item related to the testing program or surveillance data program, is that - what does that mean? Does that mean that you're just asking the applicant to provide information to demonstrate that they have a program or what does that entail? What's the expectation associated with the program?

MS. BURJA: So we are looking for plant specific data to verify that the methodology is sufficient once, you know, there are - after, you know, like, 20 effective full power years of operation, because there's, you know, sufficient conservatism for us to be able to conclude that the methodology is good for -

OPERATOR: Please pardon the interruption. Your conference contains less than three participants at this time. If you would like to continue, press star one now or the conference will be terminated.

MR. SCHULTZ: Solved that problem already. So, but is this a plant specific surveillance program?

MS. BURJA: Yes.

MR. SCHULTZ: Okay.

MS. BURJA: It's plant specific.

MR. SCHULTZ: And so you're looking for

what they're going to - what -

OPERATOR: You have activated the help menu. Press star zero -

MR. SCHULTZ: - what programs they're going to use to make sure that they have that covered?

MS. BURJA: Yes.

MR. SCHULTZ: Okay.

MS. BURJA: So we would expect to see that surveillance data and, you know, provide justification that the methodology is adequately benchmarked.

MR. SCHULTZ: In time?

MS. BURJA: Yes.

MR. SCHULTZ: Okay.

MS. BURJA: Yes.

MR. SCHULTZ: Thank you.

MS. BURJA: You're welcome. Next slide, please? So the first challenging review area that the staff encountered was that of control rod worth depletion. The staff became concerned that because full-strength CEAs with boron carbide neutron absorber can be inserted as regulating rods for various lengths of time, that boron 10 depletion could potentially affect CEA worth.

The staff discussed this issue with the applicant during an April 2015 public meeting, and issued RAIs concerning the loss of worth, potential impacts on the transient and accident analyses, and potential need for CEA service limits. In addition, the staff audited the calculation notes related to estimated boron 10 burnout.

Ultimately, the staff was able to come to a resolution on this issue because the RAI responses showed that the applicant's estimated boron 10 burnout was not negligible, but it was conservatively calculated. In addition, operating experience for the OPR1000 shows that measurements agree with predictions within the allowed uncertainty bands.

The staff also confirms that the net rod worth uncertainty listed in the DCD is much greater than the estimated loss of worth provided in the RAI response, so we have some assurance that the rod worth uncertainty would bound the loss of worth. In addition, the staff confirmed that the shutdown reactivity curve presented in the DCD was conservatively calculated.

The staff also noted that the 10-year CEA lifetime, as well as the power dependent insertion limits in tech specs helped to limit the loss of worth, and ultimately, startup physics tests will confirm that rod worth is consistent with the predictions, and if it's not, startup cannot commence. Next slide, please?

MR. SCHULTZ: Alex, the startup physics testing that you're describing here, is that for just initial startup or for startups following refueling?

MS. BURJA: It's for initial startup, and then plant procedures typically require the rod worth confirmation for every startup.

MR. SCHULTZ: Understood, thank you.

MS. BURJA: So the next challenging review area was that of the benchmarking of the nuclear design methodology. The staff became concerned that the DCD didn't adequately describe how the DIT and ROCS codes are benchmarked against experimental data that would be specific to the APR1400 nuclear design, including the use of gadolinia fuel.

So again, this was an issue that the staff discussed with the applicant during an April 2015 public meeting, and the staff issued an RAI on the topic, and audited calculation notes showing how bias and uncertainty are determined for the DIT and ROCS codes.

The staff resolved this issue because the RAI response showed that the bias and uncertainty manual is derived from measured versus predicted data for eight U.S. combustion engineering reactors that have core and nuclear designs that bound the APR1400 design.

In addition, the staff's audit of calculation notes confirmed that the methods used to determine the bias and bias uncertainties are adequate. The staff also confirmed through an audit of the bias and uncertainty manual that the high-level information about bias and uncertainty in the DCD is supported.

The RAI response also noted that the APR1400 bias and uncertainty manual is almost identical to that for OPR1000, and the staff confirmed that the OPR1000 DIT and ROCS predictions provided in the RAI response compare well against plant measurements which provides a level of confidence that the same would be true for APR1400.

The final challenging review area for the staff was that of the nuclear data that Jose brought up a little bit earlier. As he mentioned, the design methodology uses the ENDF/B-IV cross-section library that was published in 1974, and there have been many refinements and improvements to the library since then.

To assess the impacts of using the ENDF/B-IV cross-section library compared to a newer version of the library, the staff performed confirmatory criticality calculations using the SCALE code and found that the differences were substantial. If we could go to the next slide, please?

So on this slide we have the results of the staff confirmatory calculations. I ran calculations for two cases, one with a well-known criticality benchmark from the International Handbook of Evaluated Criticality Safety Benchmark Experiments, and one using a representative APR1400 3.14 weight percent on poison fuel assembly.

And in both cases, it is demonstrated that the older cross-section libraries ENDV/B-IV and even B-5, underestimated reactivity compared to the current library. If we could go back to the previous slide?

So therefore, the staff became concerned that using ENDV/B-IV could lead to inaccurate nuclear design predictions and effect transient and accident analyses as a result.

But through the staff's audit and through reviewing the RAI response, the staff came to determine that the bias and uncertainty values applied to the DIT and ROCS codes are associated with the use of ENDV/B-IV library itself, and therefore the effects of using the library are implicitly captured in the bias and uncertainty that's applied to the nuclear design calculations.

MEMBER MARCH-LEUBA: So this is a bias you applied to the k-effective?

MS. BURJA: That's correct.

MEMBER MARCH-LEUBA: So how does it affect the reactivity coefficients?

MS. BURJA: Right, so the reactivity coefficient bias and bias uncertainties are also calculated in, you know, the same manner as all of the rest nuclear design bias and bias uncertainties, so because you're already accounting for the ENDV/B-IV in those.

MEMBER MARCH-LEUBA: The biases I'm more familiar with are done on a steady state, and what was your predicted k-effective when you were on K1? Well, I think they want to say something.

MS. BURJA: Okay.

MR. DO: I'm Manseok Do from KEPCO Nuclear Fuel. You asked about a reactivity question?

MEMBER MARCH-LEUBA: I was asking, I mean, since the issue of biases because of the use of the ENDV/B-IV on k-effective, there are some biases. Is there an effect - the reliability of the k-effective is to monitor coefficients.

MR. DO: The reactivity question to -- temperature coefficients are measured for each cycle, and we compare the predictive values to measure the values. The predictive values included biases, and we are assured the biases is of proper to use for APR1400.

MEMBER MARCH-LEUBA: Sorry, I didn't follow that. So some plants do a moderated temperature coefficient experimentally by setting up two different temperatures? Is that what you did to evaluate the bias?

So sometimes the moderator temperature coefficient is evaluated by increasing the inner temperature of the core, so you operate at one temperature, and that temperature plus, say, 20 degrees, and calculate the difference in k-effective, so you experimentally measure the temperature coefficient.

MR. DO: Yes, yes.

MEMBER MARCH-LEUBA: And then you compare it to the one that you calculate, and is that how you set up the biases?

MR. DO: Yes, we measure, but we can't measure MTC. You measure ITC and we compare.

MEMBER MARCH-LEUBA: So you have a set of biases based on experimental data that you apply to your calculations?

MR. DO: Yes.

MEMBER MARCH-LEUBA: Okay, thank you.

MS. BURJA: Okay, are there any other questions on this slide? If not, we can move to the next, or the conclusion, please. Thank you.

So in conclusion, the staff was able to find that the APR1400 nuclear design meets GDC 10 through 13, 20, and 25 through 28 respectively because of the representative power distributions and limits, the instrumentation to monitor the power distributions, and the methods to control them provide reasonable assurance that the specified acceptable fuel design limits or SAFDLs will not be exceeded during normal operation or anticipated operational occurrences or AOOs.

In addition, there's a negative power coefficient of reactivity. Also, only axial xenon oscillations are possible, and can be suppressed using CEAs. In addition, there are instrumentation and controls to maintain power distributions within limits, also control systems and set points are adequate to shut down the reactor at any time.

In addition, SAFDLs will not be exceeded for any single reactivity control system malfunction as shown by the Chapter 15 transient and accident analyses and how the nuclear design was used in those. In addition, worth calculation for Chapter 4.3 demonstrate that there is sufficient shutdown margin assuming a stuck rod. Finally, there are appropriate limits on reactivity insertion rates, and appropriate reactivity values were used in the transient and accident analyses.

In addition to these findings against the GDC, the staff was also able to conclude that the applicant's analytical methods and choice of data are acceptable, and the methods are benchmarked appropriately, that the initial test program is adequate to verify the nuclear design, that the tech specs are consistent with the requirements identified in the Chapter 15 analyses to help ensure the health and safety of the public, and in addition, the staff confirmatory analyses support the foregoing conclusions.

Are there any questions at this time?

MR. SCHULTZ: Just perhaps a fine point, Alex. In that last statement for additional considerations where you say analytical methods and data are acceptable and benchmarked appropriately, would one also include the bias and uncertainty -

MS. BURJA: Yes, that -

MR. SCHULTZ: - as part of that statement that they are acceptable -

MS. BURJA: Yes, that's definitely -

MR. SCHULTZ: - because of appropriate application of that?

MS. BURJA: That's definitely part of the methodology. I would agree.

MR. SCHULTZ: Okay, that's what you considered, that it's part of the overall methodology?

MS. BURJA: Mm-mmm.

MR. SCHULTZ: Thank you.

MS. BURJA: You're welcome. If there are no further questions, I will turn it over to

Andrew. Oh, yes?

MEMBER KIRCHNER: Just a quick question. So you already looked at the reactivity insertion accidents, the Chapter 15?

MS. BURJA: I didn't personally review them, but I ensured that the initial conditions and assumptions are consistent with or are conservative relative to the nuclear design in 4.3.

MEMBER KIRCHNER: Okay, thank you. We'll get change to see 15 at some point.

MS. BURJA: That's correct.

MEMBER KIRCHNER: Thank you.

MS. BURJA: You're welcome.

MEMBER REMPE: Out of curiosity, when I was reading your draft SC or the staff's draft SC, they had a statement in there about, "Well, we know the values they assumed are typical," and then there's a little caveat about, "If it significantly differs, we'd have to evaluate," and I was just curious what is significantly different? Is that well understood in the community of folks who do this? Is it 10 percent? Is it 50 percent? What's significantly different?

MS. BURJA: I don't believe there is a hard and fast rule, and I'm trying to remember what the context of this was.

MEMBER REMPE: It was the nuclear design parameters. It was on page 4-12, fuel assemblies and core loading patterns.

MS. BURJA: Oh, right, so the DCD presents representative core loading patterns, and that's what their nuclear design calculations are based on, and the problem is if they significantly change loading patterns or, you know, any of that sort of information, it would affect the power distributions and whatnot.

MEMBER REMPE: But where is significant? Does everybody kind of agree on that and it's well understood when somebody comes in later that wants to buy this plant, and it's a certified plant, will everybody know what that means?

MS. BURJA: So I believe the information about changing the core loading is contained in the core operating limits report.

MEMBER REMPE: Okay.

MS. BURJA: So that would -

MEMBER REMPE: Give the guidance on it, okay.

MEMBER MARCH-LEUBA: My understanding of the process for the DCD is that there's a comment on the applicant to provide the first cycle and another cycle that will work, and exercise of other methods against that code. Then the real cycles after you build the plant, you analyze them one by one.

MEMBER REMPE: Yeah, okay, just wondering if it's well understood.

MEMBER MARCH-LEUBA: And the real cycle will not be the one they analyze for the DCD. It would be hopefully closed, but they will analyze it and make sure it works.

MEMBER REMPE: Okay.

MS. BURJA: Thank you. All right, if there are no further questions, I'll turn it over to Andrew Bielen from research to talk about the confirmatory analyses.

MR. BIELEN: Hi, I'm Andy Bielen. I'm from the Office of Research. I performed the initial cycle core physics confirmatory analysis, and this was to confirm the information that was presented in Section 4.3, and also to provide our basis for our cycle dependent transient and accident confirmatory analysis which we'll present to you next month.

So I want to keep today focused on the core physics itself. You know, some of that will inevitably bleed over into Chapter 15. I'll do the best I can to address that at this point in time, but I make no promises about how thorough it will be.

So what I'll talk about today is the nuclear design methods that we use here at the NRC. I'll talk about our power distribution comparisons with the initial cycle presented in DCD Section 4.3, both radial and axial.

I'll talk about the boron letdown curve as the PWR operates with soluble boron in the coolant, and that as the core depletes, that level is diluted until you have basically zero boron at the end, so we'll compare what our codes say with what was presented in the DCD.

I'll talk about the control rod bank worths and our comparisons there. I'll also just present a few safety analysis related parameters like the trip reactivity insertion curves, reactivity feedback coefficients, and then we'll just provide a summary and conclusions there.

So at NRC currently for PWRs, we're using the POLARIS code which is a method of characteristics codes relatively new to the SCALE package that Oak Ridge develops.

For this set of calculations, we used seven history conditions which the lattices were depleted, and then within those seven conditions, we have a total of 81 different branch conditions to capture different instantaneous conditions.

The reason we used so many was that we wanted to have the ability, should it be necessary, to calculate anything from an overpower condition down to cold conditions. So I think that we have confidence that our cross-section set is sufficient to carry anything that could be reasonably expected to occur within this plant.

To do the nodal core calculation, we used our PARCS code. So in the axial direction, we used 27 nodes, one reflector node in the bottom, one reflector node at the top, and then 25 in the active fuel region.

We used one radial node per fuel assembly, and 250 MWd/MTU exposure increments, so out to the full length of cycle one. Then to provide our thermal-hydraulic conditions during the core simulation, we used the PATHS code which is sort of a TH solver subcode of PARCS, and again, this is a one to one TH to neutron nodal mapping.

So first, I'll just present some of our

power distribution results. These are sort of busy figures, but this is the BOC no xenon case. We see very good general agreement here. Next slide, please?

EOC, so we're kind of tracking which assemblies are hot and which assemblies are cold versus what the applicant presented. I'd say the biggest differences are usually in the periphery, and that may be due to reflector modeling.

You know, everybody - reflector modeling is more of an art than it is a science. Everybody does things slightly differently with slightly different assumptions, but in general, I think we were pretty happy with how the radial power distribution comparisons came out. Okay, next slide, please? I'm sorry?

CHAIRMAN BALLINGER: What constitutes not good? In other words, how much of a difference -

MR. BIELEN: I'll talk about that in a subsequent slide.

CHAIRMAN BALLINGER: Okay.

MR. BIELEN: Generally speaking though just to give you a head's up, you know, when we look at - you know, we do some benchmarking with PARCS as well. You know, we have our own internal benchmarks. We have different plants that have gone through Hatch, for example, TMI, and when you look at the -

And we had an international effort a few years ago where one of the, I think it was a Norwegian country, one of the organizations there compared against some operating data from plants there, so I think that we feel good as within sort of the differences with those benchmarking efforts.

I would say, you know, in our case, the radial power distributions were all - worst-case, the RMS difference in the initial cycle, or for the initial point for the initial cycle was around five percent. It got better as the GAD burned out. The power distribution moved around, flattened out, so by end of cycle, we were at, you know, two percent RMS difference.

And in the peak power assembly, we were at, generally speaking, less than one percent different from the DCD calculation. So we feel like that is a justifiably good, you know, number.

Now, if I ran a calculation and I saw, you know, the hot assembly was somewhere completely different, or you had an RMS different of 15 percent, you know, that would be something that would cause for concern, but I think that where we were at, given the information we had available and the approaches that we used, we felt strongly that it was sufficient.

MR. SCHULTZ: The radial distribution looks at least that close.

MR. BIELEN: Yes.

MR. SCHULTZ: And, you know, I would say that was pretty good, to quote you.

MR. BIELEN: This is the axial snapshots at the beginning and end of cycle, and I think again, worst case, we're about 10 percent in the peak power node.

If you look at, you know, as Alex pointed out, you look at what the operating limits are and what the tech spec limits are versus what some of the values that we have and the differences that we have between their analysis and our analysis, there's enough margin in the design I think that any difference we have here or any uncertainties we have are well within, you know, the safety limits basically.

Okay, next slide, please? So as I said, we're happy with the radial power. We're happy with the axial power. The differences we saw are consistent with some of the benchmarking activities we've done with PARCS, so -

MEMBER MARCH-LEUBA: So yes, going back, do you use seven for your -

MR. BIELEN: Yes, our - so POLARIS, the default library is 7.1, so that is -

MEMBER MARCH-LEUBA: That could explain the difference in the two calculations.

MR. BIELEN: Yes, and in fact, you know, here when we talk about boron letdown, I think we can have a little bit of - it should be more of an interesting discussion, I think.

MEMBER MARCH-LEUBA: But on the calculations, if you have the same cross-sections and the same reflector, you should not be five percent off.

MR. BIELEN: Right.

MEMBER MARCH-LEUBA: You should be 0.005.

MR. BIELEN: Exactly, right. So some of these difference are either -

MEMBER MARCH-LEUBA: Well, four versus seven, that will do it.

MR. BIELEN: Right, right, right.

MEMBER MARCH-LEUBA: And at the end of the day, and maybe this is a question for KHNP more, you compare your power distribution results versus measured data in the plan would give you an uncertainty that I assume gets rolled over to the DNBR acceptability ratio. Is that correct?

I mean, when you calculate your power distribution, there is an uncertainty to that, and you developed the uncertainty by comparing versus measured data. That becomes your uncertainty in the power distribution, and I would put that on the acceptable value of the DNBR which was 2.49?

MR. Y. KIM: Yes, this Yun Ho Kim from KHNP. We used every - yeah every month, we measured at the site, and we compared it to the power distribution, and we got the predicted value, and if there is a reading sum limit, they are reading some limit, we just assume the current loading is okay, the current design is the same as the -

MEMBER MARCH-LEUBA: I mean, they're not within the limits? If they're outside the limits, you did something wrong?

MR. Y. KIM: I cannot remember. We would have to prepare you, but it is less than one percent or two percent.

MEMBER MARCH-LEUBA: Okay, thank you.

MR. BIELEN: Okay, so power distribution looks good. As I said, the axial offset and, you know, the other tech spec limits were well within those things. From a safety standpoint, the initial cycle is well within, you know, their operating box basically. Next slide, please?

So boron letdown, I'm going to go old school here with some arrows on some figures. On the left side, we have the critical boron concentration, and on the right side is the difference between the staff and the applicant's predictions, and I would say qualitatively, we have agreement here.

We actually have a little hump here around two or three gigawatts days per MTU where, you know, you kind of see that as like the gadolinia burnout, and then the core starts depleting across the board.

Now, the big difference here is around three to four out to around 12 gigawatt days per MTU. Now, like when you do - when plants operate and they measure what their critical boron concentration is against what their predictions are, generally speaking, the acceptance criteria is around 50 ppm, you know, so if you're within ppm, you fell okay that your prediction looks good. Now, here we do have an example where we're at worst case about 125 ppm difference between our calculation and what the simulation of the applicant is.

Now, you know, and I think that this is a case, when you look at neutronics and you look at things like what the cross-sections are doing, what the core condition is from a thermal-hydraulic standpoint and all of these things, I think the most likely explanation here is differences in probably gadolinia treatment as the core burns out.

MEMBER MARCH-LEUBA: So is this the staff's depletion calculation?

MR. BIELEN: Yes, this is our - I'm sorry if I didn't make that clear. Yes, this is our depletion calculation.

MEMBER MARCH-LEUBA: But you're using different cross-sections than they are?

MR. BIELEN: Right, like I said, our cross-sections are NDF B7.1. Theirs are NDF 4. However, you know, they do have their, you know, benchmarking that they use from operating plants where they establish the biases and uncertainties. You know, we don't have that level of benchmarking available to us at this point because we're not actually running a plant ourselves.

So it's possible that, you know, whatever differences that they see, they've captured within their biases and uncertainties. I'm not really sure how, you know, the values that they present here compare with whatever raw values come out of their codes.

So next slide, please? So like I said, qualitatively very good, and like I said, 50 ppm is typically sort of the number I look at where I start worrying that the differences are significant, and I think that it's an isotopics issue. It may be an TH issue as well. You know, I don't know anything about what the TH solver is in ROCS. I doubt it, but it's a possibility.

Now, you know, one thing that makes us feel good about this is that the biggest differences are in the middle of the cycle. However, from a reactivity standpoint or a reactivity feedback coefficient standpoint, your limiting values are always at either beginning of cycle or end of cycle. You know, you're either most positive or most negative at those points.

So if we have bigger differences in the

middle of the cycle, a, from a safety standpoint, those wouldn't be limiting times in the cycle for almost all of the transient analysis in Chapter 15, and b, they monitor boron concentration at the plant every single day, of course. It's part of how the plant operates.

If they're operating the plant and they see, oh, geez, I've exceeded my predicted critical boron concentration acceptance criteria, they would have to go into their tech specs, you know, reevaluate, and figure out what's going on. So there are big differences here.

I don't feel like it's a safety concern. It is something that, you know, we as the staff will probably look at PARCS, you know, and sort of assess, you know, where we think the issue is.

MEMBER MARCH-LEUBA: When I look at slide 22, the previous slide, of course I just looked at it, it feels to me that you're depleting two isotopes at a different rate than they are.

MR. BIELEN: Okay.

MEMBER MARCH-LEUBA: You have the beginning one that you depleted too fast, and another one which is slower that you depleted too slow.

MR. BIELEN: Okay.

MEMBER MARCH-LEUBA: It feels like - I mean, that's why you have that shape. One is going down and the other one is going up, and then when you sum them up, you get that.

MR. BIELEN: Okay.

MEMBER MARCH-LEUBA: So in my previous life, depletion with PARCS, there isn't that much benchmark.

MR. BIELEN: Right, that's, I mean -

MEMBER MARCH-LEUBA: It's a new - I mean, we're not even able to do it. It's thanks to PATHS we're able to do it now.

MR. BIELEN: Right.

MEMBER MARCH-LEUBA: So I would go with the blue line as the more accurate.

MR. BIELEN: Right, and as I said, you know, we do have benchmarking data with PARCS. I would definitely say that the applicant, by virtue of operating these plants, you know, has a much bigger data set available to them to verify their codes.

MEMBER MARCH-LEUBA: And the depletion with PATHS, I know we've done it for one of the foreign reactors and it worked very well, but there isn't as much experience.

MR. BIELEN: Right, absolutely.

MR. SCHULTZ: I agree with your conclusion that with regard to safety analysis, the conclusion is certainly valid and appropriate. At the same time, operationally, this would present a concern even though you can certainly modify the concentrations and move through operations if you had this consideration, but has the applicant -

It would prompt me to get more information about the applicant's experience with their codes for their plants for their operational experience to ensure that their expectation is not to see this difference -

MR. BIELEN: Right.

MR. SCHULTZ: - when they use their tools, which they have proposed to use -

MR. BIELEN: Right.

MR. SCHULTZ: - in doing the analysis cycle by cycle.

MR. BIELEN: Right.

MR. SCHULTZ: Have you done that to get some more information about their experience base?

MR. BIELEN: I mean, I think I would, you know - personally, no. The NRO staff that did the review appeared to do a really good job of looking at what the applicant's biases and uncertainties were, you know, using DITS and ROCS.

I think that that review work or that review effort there would have picked up and, you know, you can speak to this better than I can, but, you know, this seems like a large chunk of, you know, what you would be concerned about, that I think that those would be picked up within, you know, the biases and uncertainties manual that the applicant made available to us.

MR. SCHULTZ: Would you like to comment on that?

MS. BURJA: Yeah, I would agree with Andy on that.

MR. SCHULTZ: Okay, I appreciate that. Thank you.

CHAIRMAN BALLINGER: The good news is that we were ahead of schedule and so we were able to accelerate things. The bad news is now we have to decide on when to break. There are a number of meetings that are happening at noon, and so I guess we need to decide what is a good breaking point. I'm looking at the slides and it doesn't look like it's a great point here, but I'll leave that up to you.

MR. BIELEN: I mean, I'm good to go. You guys can -

CHAIRMAN BALLINGER: What does that mean? You're here until nine?

MR. BIELEN: I am here at your pleasure, sir.

CHAIRMAN BALLINGER: That doesn't answer the question.

MEMBER MARCH-LEUBA: There are four slides on 4.3. Do you think maybe you can go in seven minutes?

MR. BIELEN: Yeah, I can certainly present quickly, you know. It depends on how many questions you want to ask me.

CHAIRMAN BALLINGER: All right, well, let's push on and see what we can do, but we do have, I think we have PNP at noon and another meeting, so there's a fairly hard point that we have to deal with.

MR. BIELEN: Okay, well, I can get through these quickly if you let me.

MEMBER SUNSERI: Before you move on, I just had one more question about the boron letdown curve. You know, essentially, that's a measure of the energy that's in the core, and you were talking about conservative values were used. I presume you looked at this and said, "Well, we're predicting less energy than they are, so that's conservative." Is that -

MR. BIELEN: Yeah, I mean, I think that that's - I would put it that because they're predicting a higher boron concentration, they see the core as more reactive than we do, you know. I have a hard time in general when you talk about nuclear design saying conservative versus, you know, non-conservative because, you know, it's a value problem, you know, so high one place means low another place, you know, but if you want to put it that way, then, yes.

MEMBER SUNSERI: Yeah, I didn't frame my question very well. I'm talking about the handoff back to the safety analysis.

MR. BIELEN: Yeah, and I think that again, because your extreme values of boron concentration are still pinned at the beginning and end regardless of what the path is in between the two, that you're still covered in your safety analysis by what you see in the core design.

MEMBER SUNSERI: Okay, thank you.

MR. BIELEN: Sure. Okay, so there are a bunch of control rod banks. We looked at the worth of every single one of them and compared them against, and these are differences in worth versus the applicant.

And when I say present, I mean if they present a calculated 10 percent delta K over K for a given rod bank worth and we calculated 11 percent delta K over K, then the difference in worth is then just 10 percent. Ten percent is again sort of the figure of merit that I think of when I think of control rod bank worths.

In general, we met that metric pretty well, I think. And given the fact that in the safety analysis, aside from your highest worth of stuck rod, everything goes in, you know - I think that we're doing pretty well as far as bank worth is concerned. Next slide, please?

So this is our comparison of what the trip reactivity insertion will look like at different core conditions versus what they use in the DCD safety analysis.

It's just present - this is a normalized curve, so percent rod insertion from the top of the core versus what the available worth is, or what worth is being put in at that given position, and so I think that our calculations indicate that whatever the DCD is using is pretty representative of what you would expect in an actual cycle.

So talking about feedback coefficients here, so when you have a fresh core that has no burnable poison, so initial criticality basically, or I'm sorry, no xenon/samarium, you have a limiting point here where your moderator density coefficient is very slightly positive, and that's actually not right. That's 9.7 times 10 to the minus four dollars per M cubed kilogram, and your fuel temperature coefficient. So the MDC being positive means the MTC is negative. Yeah, let me think about this. I'm sorry.

MEMBER MARCH-LEUBA: Temperature goes up, density goes down.

MR. BIELEN: Right, right, right. Anyway, when you put these - so you have a point here where you may have a positive MDC, or a negative MDC and a positive MTC.

However, the fuel temperature coefficient is strong enough of a contributor between those two things that the overall power coefficient is still negative as we indicated from our 4.3 review, and I think that we're, from a safety standpoint, we're still where we want to be.

And then I think that we have run some point kinetics calculations on the initial core with, you know, coupled with TRACE, so we can talk a little bit more about that at our Chapter 15 meeting I think.

So next slide? All right, so power distribution we feel looks really good, or at least, you know, we feel good about them. The peak power assemblies were within about a percent, which is reasonable to excellent. We feel like the agreement in the axial power distribution is also good, and we're far enough away from any tech spec limits that were covered by safety analysis.

Boron letdown, good qualitative agreement, some disagreement probably due to isotopic or treatment of gadolinia, and Jose, I'll bring that back to the PARCS developers and see if we can - if it is indeed a PARCS issue, it's something that we should try to resolve.

MEMBER MARCH-LEUBA: I would suspect it's not a PARCS issue, but it was an -- issue.

MR. BIELEN: Okay, all right, we'll see. Our predicted bank worths were in good agreement, and as was the reactor trip characteristics, and we think that, you know, the initial cycle, based on our confirmatory calcs, is very reasonably within the safety box that is spelled out by the tech specs and the accident analysis, so the initial cycle should not challenge any of the safety analysis. So, I think that's all I have.

CHAIRMAN BALLINGER: You're well within the margin of error, including bias and uncertainty. Thank you very much. Any questions from the members before we break for lunch? In that case, we will recess until 1:00 p.m.

(Whereupon, the above-entitled matter went off the record at 11:56 a.m. and resumed at 1:01 p.m.)

CHAIRMAN BALLINGER: Okay, we're back in session. The floor is yours.

MR. WUNDER: Thank you, Mr. Chairman. I'm George Wunder and I am joined now by Carl Thurston and Jim Gilmore, who will be doing the Staff presentation on Section 4.4. Carl, please?

MR. THURSTON: Okay.

CHAIRMAN BALLINGER: Same rules apply as this morning.

MR. THURSTON: Didn't see it. My name is Carl Thurston and I'm a recent transfer from the

Office of Research and I will present the Staff's evaluation of Section 4.4, Thermal and Hydraulic Design.

Since the RCS design and configuration is essentially that of the System 80+, the review is primarily focused on confirming the applicability of previously approved CE methods and the Applicant's Technical Reports.

So, first of all, the key areas that we reviewed included thermal and hydraulic design of the RCS, the core in RCS, confirming that acceptable analytical methods were used. Largely, the methods were largely based on System 80 and System 80+ data, confirming acceptable margins against conditions leading to fuel damage during normal operation and AOOs and confirming that the design is not susceptible to thermal-hydraulic instability.

The key areas also included calculated core parameters to establish minimum DNBR hydraulic loads on the core and RCS during normal operation and DBA conditions. The loads were extrapolated from System 80 flow data, primarily Palo Verde. We also confirmed negligible differences in APR1400 core geometry and operating parameters, as compared to System 80+ and System 80 designs.

CHAIRMAN BALLINGER: Back to the last point, negligible differences between System 80+ and System 80, have there been measured core bypass numbers for those cores?

MR. THURSTON: The core bypass flow is the same in the DCD. Jim, if you have any?

MR. GILMER: As far as I've seen, I don't think even Palo Verde has measured the bypass.

CHAIRMAN BALLINGER: Okay. Just curious as whether something was actually done on it.

MR. GILMER: We'll take that up with our colleagues in NRO to --

MR. LU: Shanlai Lu, from Staff, so, even for System 80, 80+, at Palo Verde was not measurable.

CHAIRMAN BALLINGER: Okay.

MEMBER MARCH-LEUBA: So, let's follow-up on that. The assumption then is that this source that does not have really significant impact on the worth measuring, I guess. For the bypass flow.

CHAIRMAN BALLINGER: Member Skillman showed me some calculations where, if you're off by a few thousandths of an inch, you can make a huge difference in bypass flow.

MEMBER MARCH-LEUBA: Yes. If it was my

plant, I would like to measure it.

MR. LU: Okay. I think the way they approach the bypass flow is really -- oh, sorry. Here? Okay. The way they approach the bypass flow, by itself, you look at how to calculate the resistance itself.

It's really, the core condition, calculating can be different. If you have thermal expansion, you have different. So, all those bypass flows in terms of percentage wise are so small at this point, it's considered as part of the uncertainty.

CHAIRMAN BALLINGER: Okay.

MR. LU: All right.

MEMBER MARCH-LEUBA: That is a good answer.

MR. LU: That's part of uncertainty.

MEMBER MARCH-LEUBA: It's basically what I was saying before --

MR. LU: Right.

MEMBER MARCH-LEUBA: -- but in a more coherent way.

MR. LU: That's right.

MEMBER MARCH-LEUBA: That you can bury it into the uncertainties.

MR. LU: That's right.

CHAIRMAN BALLINGER: Okay.

MR. LU: So, I don't want to take Alex -- any other questions about the bypass before we go forward? I'm just trying to -- if no other questions, then I'm off the table.

MR. WUNDER: Thank you for that. Okay.

MR. THURSTON: So, we're continuing with the areas of review. So, the areas of review also included uncertainty analysis methodologies, namely Statistical Combination of Uncertainties, to ensure that at least 95/95 confidence level that the hot fuel rod does not experience DNBR during normal operations, AOOs.

CPCS, core protection calculator system, interfaces that support DNBR and local power density safety limits. COLSS, core operating limits supervisory system, interfaces with the CPCS that support Chapter 7 reviews.

And computation of CPCS parameters needed for core reload. Staff also noted that reactor protection system design and operation, namely the COLSS and CPCS, is essentially that of the Palo Verde System 80.

So, the Technical Reports we reviewed

included Thermal Design Methodology. This report described overall CE methodology based on approved TORC and CETOP codes, with the KCE1 CHF correlation and CE methodology for statistical uncertainties methods used to ensure 95/95 confidence level that hot fuel rod does not experience DNBR during normal operations or AOOs, consistent with SRP, including a penalty for rod bow.

The Technical Reports also included Functional Design Requirements for CPCS for the AP1400 and Functional Design for the COLSS. These described CE methods based on approved methodology used at Palo Verde, but also implemented at San Onofre, ANO-2, and Waterford.

The CPCS protection software design assures 95/95 confidence that DNBR and LPD limits are maintained. Procedures for development of CPC constants will be developed by the COL holder, consistent with existing procedures for operating System 80 plants.

The next Technical Report was CPC Setpoint Analysis. This report describes how CPCS computes changes in linear power density and describes measurements of core conditions for peak power density and DNBR based on ex-core instrument measurements.

And CPC uses monitored parameters to calculate LPD and DNBR margin to trip limits. Cycle-dependent uncertainties associated with CPC trip point settings are combined such that the adjusted LPD and DNBR setpoints are always conservative.

The next Technical Report is Uncertainty Methodology and Application for Instrumentation. This report describes methodology used to combine uncertainties to ensure that plant protective functions activate at desired values under normal and accident conditions, and is essentially the same as the CE Topical Report CEN-356 approved by Staff in 1988.

The next Technical Report is Setpoint Methodology for Plant Protection System. It describes methodologies used to establish setpoints to be used for ex-core plant protection system trip settings. PPS functions contain the other RPS trips and the ESFAS trips.

Setpoints established such that during DBEs, the analytical limit is not exceed and the analytical limits are established such that safety limits are not reached and the safety limits assure that unacceptable consequences do not occur during Design Basis Events.

So, I'll next review some of the challenging areas of review that we encountered. This one is concerning the CPCS and COLSS functions and interfaces. The basis for the CPCS was not well documented in the System 80+ DCD.

KHNP followed the System 80+ DCD, but did not link references to post-1980 functional changes and improvements to the current System 80 design. So, we had a concern that the documentation was not adequate to assure -- for us to make a safety finding that the design was correct and safe.

The Staff conducted an audit in January 2016 to address the issue. The issue was ultimately resolved by finding the CE references and confirming that the functionality are based on approved and implemented changes at Palo Verde.

MR. SCHULTZ: Carl, just a moment there. This goes back to a slide that was, I guess it's Slide 31, which states that the procedures for the CPCS constants were developed using existing procedures for the System 80 plants, existing procedures. This seems to suggest -- so, that's what -- so, now, you're saying that you needed to go back and validate that?

MR. THURSTON: No, we're saying that the COL holder will need to develop procedures similar to the procedures that already exist for existing System 80 plants.

MR. SCHULTZ: Similar to?

MR. THURSTON: Similar to.

MR. SCHULTZ: Okay.

MR. THURSTON: That's the intent.

MR. SCHULTZ: All right. But, I guess my question is, is there sufficient guidance for the COL to do that?

MR. THURSTON: Jim?

MR. GILMER: There's really -- there is some guidance in, I believe, CEN-201 and --

MR. SCHULTZ: Okay.

MR. GILMER: -- I don't remember, but --

MR. SCHULTZ: That sounds like the right number to me, but go ahead.

MR. GILMER: Yes. And we looked at the existing procedures for Palo Verde, as well as the recently commercial Shikori 3. KHNP routed --

MR. SCHULTZ: Okay.

MR. GILMER: -- that to us. So, we compared that to Palo Verde's procedures and felt that it was adequate. But we just ask that they include a COL holder item to develop the procedures.

MR. SCHULTZ: To make sure that that was done by the holder?

MR. GILMER: Correct.

MR. SCHULTZ: Okay. Thank you. I understand. But you reviewed that that had been done previously --

MR. GILMER: Correct.

MR. SCHULTZ: -- so, it should be appropriately applicable going forward?

MR. THURSTON: That's right.

MR. GILMER: Correct. For ANO-2, San Onofre, and Palo Verde.

MR. SCHULTZ: So, what you did was prescribe that in the future, it would be, what I suggested needed to be done would be done?

MR. GILMER: That is correct.

MR. SCHULTZ: All right. Thank you.

MR. THURSTON: Okay. So, the next area of challenging review involved single-sided versus double-sided confidence. The Applicant used a Statistical Combination of Uncertainties methodology and we reviewed that methodology to confirm that the DNBR safety limit of 1.124 being converted to the analytical limit of 1.29 was correct with the 95/95 confidence level.

And we found out that, basically, they had used a previous Revision of 1.125, or, excuse me, 1.105, which is the Reg Guide. And that Reg Guide had been updated since the CE methodology was approved and KHNP was following the previous methodology that CE had used prior to this Reg Guide being updated in 1999.

So, single-sided methodology was based on the CEN-356 approved by the Staff in 1989, but, again, that's prior to the change in the Reg Guide in 1999. So, there -- and there's some other people that can speak to this. Joe?

MR. ASHCRAFT: Joe Ashcraft.

MR. THURSTON: Yes, it's a Chapter 7, really, item that affects our Section because of the 1.29 analytical limit.

MR. ASHCRAFT: This is Joe Ashcraft, I'm NRO I&C. So, a colleague originally started this review of this methodology, along with the setpoint methodology, however, at some point, he went to a different job, so I took over.

So, there is an RAI to Chapter 7 that discusses this issue of the use of the -- they're using a conversion of 1.645 and, effectively, that just decreases the margin. So, Rev. 3, which requires the 95/95 confidence level, the Staff, NRO Staff, we did not accept that use of 1.645 ratio. So, that's the point at this time.

So, from our viewpoint, those two setpoints are safety-related and are required to be in conformance with Reg Guide 1.105 Rev. 3, as their DCD stipulates at this time. So, we're not arguing that this methodology that's been existing since 1980 is not valid, because it's an algorithm kind of thing that we punt to reactor systems.

Our RAI basically says that use of that ratio of 1.645 over a two sigma value does not meet 1.105 Rev. 3. There are some other minor issues, but that's probably the major sticking point. We have been in conversations with the Chapter 7 staff, KHNP staff on this issue, and we presented some of our questions via a setpoint audit that they're now going to resolve or answer. So, that's where we stand from a Chapter 7 perspective.

MR. THURSTON: Thank you, Joe. So, now --

MEMBER SUNSERI: So, if we go to the double-sided confidence level versus the single, what's the impact on this number then?

MR. THURSTON: It would probably go up.

MEMBER SUNSERI: Does it change it --

MR. THURSTON: It would.

MEMBER SUNSERI: -- substantially?

MR. THURSTON: I think so. We haven't crunched the numbers, it's a very complicated calculation.

MR. ASHCRAFT: This is Joe Ashcraft again. Just from our experience, from setpoint arena, not necessarily this methodology, but when you use that single-sided ratio, it impacts the margin between your safety limit or analytical limit and your LSSS setpoint by approximately 18 percent.

MR. THURSTON: So, quite a bit.

MR. SCHULTZ: So, Joe, the -- I'm sorry to pull you back, it should be a simple answer, because it's just a repeat-back so I understand. The expectation, then, is that there will be a clarification from the Applicant to identify clearly what Revision they are performing the analysis to, the single-sided confidence levels approach, and if they wanted to take advantage of Rev. 3, they would have to come back? Come back for approval?

MR. ASHCRAFT: Right. So, from Chapter 7

staff of KHNP that we've talked to, we more or less presented -- so they could continue to say they meet 1.105 Rev. 3, which they would have to come back with more information.

However, a path forward that we presented that could work as well, since this methodology has been in use since the 1980s, would be to take exception for these two setpoints and denote in Chapter 1 that these two setpoints do not meet 1.105 Rev. 3 and that they meet the regulations per this methodology that Chapter 4 Staff has reviewed. And there's a few minor changes in Chapter 15 and Chapter 7 that would have to take place, but then, we would have no issue.

MR. SCHULTZ: So, you're clarifying to me that they have, they still have two paths to reach resolution, but they've got to choose one and go forward?

MR. ASHCRAFT: Exactly. And that's how we presented it to the Chapter 7 staff. I don't know if Chapter 4 staff was made aware of that conversation, which happened last week.

MR. SCHULTZ: Understood, thank you.

MR. THURSTON: So, we can conclude for Section 4.4, the Staff findings that, basically, the

GDC 10, the SAFDLs are not exceeded during normal operation and AOOs and, GDC 12, suppression of reactor power oscillations can be reliably detected and suppressed.

In summary, the Staff notes that the thermal design methodology depends heavily upon codes and methods previously approved by Staff for domestic CE plants. The APR1400 thermal hydraulic design is comparable to System 80+, with small differences due to the slight increase in power, less than two percent. And the Staff concludes that the design provides adequate assurance that the reactor will perform its related safety functions under all modes of operation, pending completion of open items.

MEMBER KIRCHNER: So, may I ask, the existing fleet of CE designs that share a lot of the same background, they're using the previous version of Reg Guide 1.105?

MR. THURSTON: Correct.

MEMBER KIRCHNER: So, you're saying that if they just make a change to reference Rev. 2, that's acceptable?

MR. THURSTON: That's one path forward.

MEMBER KIRCHNER: Okay. Thank you.

MR. THURSTON: Thank you. Jim, you --

MR. GILMER: Were there any more questions before -- I wanted to address Dr. Rempe's question from this morning on the KCE1 correlation. One of the references in the Staff SE for 4.4 is the vendor inspection of the software quality assurance. So, I have a reference for that, I can give you an ADAMS number and --

MEMBER REMPE: Please send it to Chris, who's not in the room right now.

MR. GILMER: Okay. I'll --

MEMBER REMPE: Where is Chris?

MR. GILMER: I'll send it to him.

MEMBER REMPE: Yes, please send it to him and -- yes.

MEMBER MARCH-LEUBA: He's likely in the locker room.

MEMBER REMPE: Okay. So, yes, if you'll send it to us, that would be great.

MR. GILMER: Okay. I was going to say that this inspection was focused on three codes, RELAP, CSAC-3 and CETOP-D, but in the process of the inspection, we also examined all of the records for both TORC and the older version of CETOP, as well as the implementation of the KCE1 correlation and their validation. And they basically did it identically for the CETOP-D and the CETOP.

MEMBER REMPE: Okay.

MR. GILMER: And there's a statement in the vendor inspection report that the inspection team verified the thermal margin preserved by CETOP-D is conservative with respect to analysis performed with the TORC.

MEMBER REMPE: Thank you.

MR. GILMER: So, the dots may not be most clearly connected, but --

MEMBER REMPE: But they are.

MR. GILMER: -- there is a reference.

MEMBER REMPE: Okay, thank you.

MR. GILMER: Okay. Further questions on this Section?

MR. WUNDER: Okay. I guess that's it for Section 4.4. If we could have our Section 4.5 people come up now? We've been joined by John Honcharik and Dan Widrevitz, of our Materials and Chemical Engineering Branch.

MR. HONCHARIK: Hello, my name is John Honcharik. I'm a Senior Materials Engineer in NRO Division Engineering and I reviewed the Section 4.5.1, CRDs, so today, I'll present to you the materials that were proposed in the design, based on the application and also the responses to RAIs.

As you can see here, material selection and fabrication techniques and heat treats and cleanliness control were in accordance with the NUREG-0800. The materials that were used for the pressure boundary parts of the CRD were consistent with other designs and have satisfactory operating experience.

Some of the materials used were stabilized stainless steels, martensitic stainless steels, nickel alloys, basically alloy 690 that are thermal treated, and austenitic stainless steels 304 and 316, and the associated filler metals.

For the materials for the non-pressure boundary components, they were also consistent with other designs and have very good operating experience. And they also included austenitic stainless steels 316, 321, and 204, and martensitic with certain types of heat treating and also, nickel-based alloy X-750 and alloy 625.

Now, I'll discuss the fabrication techniques proposed with the cleanliness requirements. Basically, with the austenitic stainless steel base materials, they're going to be consistent with the recommendations that are in the Reg Guide 1.44 and, basically, the procedures, including welding procedures, will be demonstrated and tested to make sure that they don't sensitize the stainless steel components.

Also, the controls for abrasive work and grinding and cleaning are going to, basically, be in accordance with criteria in Reg Guide 1.28 and ASME NQA-1. The heat treats for some of these alloys were consistent with NUREG, including the operating experience, such as alloy X-750, which is heated to at least 1,149 degrees C, and Type 410 Condition T, heat treated above 565 degrees C.

CHAIRMAN BALLINGER: I have a question here.

MR. HONCHARIK: Yes?

CHAIRMAN BALLINGER: In Chapter 4, it lists this alloy X-750 heat treatment, that's not a heat treatment, that's a solution treatment. Most of the times, X-750 and these others are followed by an age.

MR. HONCHARIK: Right.

CHAIRMAN BALLINGER: In Chapter 4, it says that the heat treatment is designed to be resistant to stress corrosion cracking, but that heat treatment -- is that true, that you're just going to use it in the solution and the conditioning?

MR. HONCHARIK: Yes. Because that might -- yes. It's basically a solution anneal.

CHAIRMAN BALLINGER: So, that's it?

MR. HONCHARIK: That's it. And not doing any other -- you're using the solution anneal.

CHAIRMAN BALLINGER: Okay. And the same thing goes for the A-276? No, the 410?

MR. HONCHARIK: Yes, from what I can remember.

CHAIRMAN BALLINGER: That's more likely the 410?

MR. HONCHARIK: Right.

CHAIRMAN BALLINGER: Okay. All right. That's an unusual heat treatment.

MR. HONCHARIK: And then, the last thing I want to talk about is the only remaining issue that we had was dealing with the venting device, that is called Versa Vent. The Versa Vent is used to vent the CRD to minimize the levels of oxygen in these dead-leg areas.

I guess, right now, the Applicant considers the Versa Vent to be a non-pressure boundary component, even though it replaces a pressure boundary component. So, the Staff considers it pressure boundary and also, the Staff requested that the material specifications and types be provided so that Staff can review what the material is for compatibility with the other materials. A response has been received and is currently in evaluation.

As part of this issue, there is another open item and it deals with the Versa Vent and whether or not there is operating experience with data that shows that venting during these refueling outages will reduce the oxygen levels, so that non-L grade stainless can be used. And they've provided a response and so far, that response has a lot of data and operating experience for that, so that looks pretty good. And that concludes my talk on CRD materials.

MR. WIDREVITZ: All right. So, can we actually start on Slide 41, please? My slides were backwards. So, off to a good start. Section 4.5.2, I am Dan Widrevitz. Section 4.5.2 is on Reactor Internals and Core Support Materials.

The Staff review focuses on the topic areas of material specifications, selection, and heat treatments, controls on welding, nondestructive examination, austenitic stainless steel issues, other material issues, and other degradation mechanisms, specifically IASCC and void swelling.

So, in terms of this review, this design is very consistent with industry practice and previous experience. So, the Applicant had a very high level of adherence to what we consider appropriate controls, such as Reg Guides on sensitization and on cleanliness.

The Staff RAIs primarily focused on clarification and completeness, getting to the same page of what needs to be in a DCD and what the Staff needed to adequately review on the topics. Certainly, we were happy with where we ended up at the end and I'm wondering, I heard some questions earlier during the KHNP section, I was wondering if you wanted to bring up anything here for me.

CHAIRMAN BALLINGER: A few of us are mystified about the use of cobalt in some of these CRDMs and stuff like that and we're just curious about that whole issue. You're comfortable with -- I mean, it's a maintenance issue, when it comes right down to it, but, yes, are you comfortable with that?

MR. WIDREVITZ: For my part, it's not any different than what we've been seeing in a lot of the existing reactors and they haven't had particular trouble. There's -- obviously, we don't spray it on every corner of the internals, but there's a couple places where you want it for hard-facing.

CHAIRMAN BALLINGER: Yes. I guess I thought differently. That there was a big effort to get cobalt out --

MR. WIDREVITZ: Yes, and there's --

CHAIRMAN BALLINGER: -- of the primary systems --

MR. WIDREVITZ: It is mostly out.

CHAIRMAN BALLINGER: -- for sure.

MR. WIDREVITZ: It's mostly out, but there's still a few places where its used.

MR. HONCHARIK: Yes, this is John Honcharik. Most -- I think Dan's right. They've tried to minimize it, but there are certain areas where they can't. So, they have to use that still, like for the hard-facing.

Sometimes, they tried to use some of the chromium oxide to do it, but in certain areas, it won't get the wear properties, so they need to use that. So, they tried to reduce it, but they can't go away from it. And other designs are similar in that respect.

CHAIRMAN BALLINGER: I recall some time ago where they had a very large program at EPRI to eliminate, to find a suitable replacement, and they failed.

MR. HONCHARIK: Right.

CHAIRMAN BALLINGER: For some applications. Thank you.

MR. SCHULTZ: Is this hard-facing on the control rods systems or is it on the guide tubes or neither?

MR. HONCHARIK: There are some small little, like, pins and stuff that are in the CRD, where they need hard-facing, and latch mechanisms. So --

MR. SCHULTZ: Oh, okay.

MR. HONCHARIK: -- other than that, I'm not sure how much --

MR. SCHULTZ: On the drive mechanisms?

MR. HONCHARIK: Yes. Most of it, I think, is in the CRDs. There might be very minimal in --

MR. WIDREVITZ: It's in similar --

MR. HONCHARIK: -- reactor internals.

MR. WIDREVITZ: -- locations in the reactor internals --

MR. HONCHARIK: Yes.

MR. WIDREVITZ: -- where there's latches.

MR. SCHULTZ: I understand. Thank you.

MS. BURJA: All right. Again, my name is Alex Burja and I will present to you the Staff's review of DCD Section 4.6, Functional Design of Reactivity Control Systems.

I'll start out by saying that in the Staff's review, there were no new or significant issues that came up, as it's a pretty industry standard design. So, I'll just go over the areas of review and then, offer conclusions, as there were really no challenging areas of review.

So, in terms of the functionality and arrangement of the control rod drive system, or CRDS, it's pretty standard. It consists of control element drive mechanism, or CEDMs, and the digital rod control system, or DRCS.

And during normal operation, the DRCS actuates the CEDMs to insert or withdraw the CEAs. Under conditions of reactor trip, the trip switch gear removes power from the DRCS, which would deenergize the CEDM coils and drop the CEAs into the core. So, fairly standard.

The Staff also reviewed the environmental and seismic qualifications of the CRDS, since the CRDS needs to remain functional during and after design basis events and be able to withstand harsh environments.

In addition, the Staff ensured that the CRDS cooling system meets the design requirements so that the CRDS can remain functional. In addition, the Staff examined possible single failures of the CRDS to ensure that no single failure will affect the essential trip function or result in violating SAFDLs.

The Staff also looked at the testing and verification for the CRDS. So, testing for the CEDMs is actually done under SRP Section 3.9.4, but in terms of this review, the Staff looked at the initial test program for the CRDS to ensure that there was enough in there to verify the functionality of the CRDS. In addition, the Staff looked at the ITAAC and tech spec surveillance requirements for scram time, which serve as experimental verification that the trip function will work.

The Staff also reviewed the combined performance of reactivity control systems. So, basically, how the CEAs work together with the soluble boron from the CVCS, as well as safety injection system to mitigate transience and postulated accidents.

And ultimately, the Chapter 15 transient and accident analyses will show that the reactivity control systems are capable of working together to control the reactivity changes during design basis events.

The Staff also reviewed possible common mode failures of the reactivity control systems. And since these systems are completely independent, the Staff identified no potential common mode failures besides the possibility of postulated pipe breaks and associated missiles. And that review is conducted under SRP Section 3.6.

Finally, the Staff reviewed tech spec requirements for the reactivity control systems, such as rod insertion and alignment limits and charging flow limits, to ensure that there are appropriate limits on reactivity control.

So, in conclusion, the Staff finds that the CRDS, in conjunction with the other reactivity control systems, meets GDC 4, 23, and 25 through 29, for a number of reasons, including the design qualification and physical protection of the CRDS provide reasonable assurance that it will remain functional and be able to safely shut down the reactor under adverse environmental conditions and after postulated accidents.

In addition, the CRDS is proven to fail in a safe condition. Also, no single malfunction in the CRDS will result in exceeding SAFDLs, as was demonstrated in failure modes and effects analyses and as will be shown in the Chapter 15 transient and accidents analyses. Also, the CRDS and CVCS are independent reactivity control systems based on different design principles and are capable of reliably controlling the rate of reactivity changes during normal operation.

MR. SCHULTZ: Alex, they're --

MS. BURJA: Yes?

MR. SCHULTZ: They're independent, so each are capable of reliably controlling the rate of reactivity changes?

MS. BURJA: That --

MR. SCHULTZ: Each are capable of?

MS. BURJA: That's true. So, both are

provided, because the CEAs do the short-term reactivity control and then, the soluble boron is there for the long-term control. But the requirements for soluble boron, for instance, are, like, being able to hold the core sub-critical under cold conditions, that's also part of the GDC. But each system is capable of performing the reactivity control requirements.

MR. SCHULTZ: Without the other? That's what I want to get to?

MS. BURJA: Not without -- so, we need, a design needs both of them.

MR. SCHULTZ: I understand that, but in terms of capability, are they independently capable of controlling the reactivity changes under normal operation?

MS. BURJA: I'm not really sure how to answer that.

MR. SCHULTZ: Okay.

MR. LU: This is Shanlai Lu, from Staff. Steve, yes, the answer is yes and they bank them together depends on the different banking and for the safety shutdown rod or whatever the control rod there. But that's related to the independence of the control system of the Chapter 7 review.

MR. SCHULTZ: Okay.

MR. LU: But the answer to your question is, yes.

MR. SCHULTZ: Thank you.

MS. BURJA: Okay. Picking up where I left off, the Chapter 15 transient and accident analyses demonstrate that the reactivity control systems can work together to control the reactivity changes during design basis events. In addition, the reactivity insertion limits in the tech specs help to or will prevent a prompt power excursion.

Finally, the design and testing of the reactivity control systems ensure an extremely high probability that they will accomplish their safety function during AOOs. In addition to the conclusions about the GDC, the Staff was also able to conclude that the CRDS cooling system meets the design requirements.

In addition, the initial test program is adequate to verify the reactivity control systems. And the tech specs related to the reactivity control systems are adequate and are consistent with the requirements that are identified from the transient and accident analyses. That concludes what I have for Section 4.6. Are there any questions at this time?

MEMBER REMPE: Okay. So, I don't have any on this Section, but I have something, just a higher level question about the -- that came up today about Section 4. And, actually, Matt started the questioning and I've been trying to follow it up. Is it okay to bring that up now or do people have questions on this? Okay.

So, earlier today, Matt mentioned the load-following question and KHNP came back with the RAI 2938332, which was issued in November 2015, and it clearly identifies the Staff's concern about the use of the word load-following and, actually, it shows up in many Sections, and in Section 4, the way that, in my opinion, which doesn't count for anything, since I'm a single member, but they addressed it fine in Section 4, but in Section 3, I'm not so happy with it, and it doesn't even mention Section 10, or Chapter 10.

And I'm just wondering what the Staff's position is. And the reason I'm asking it is, to me, it would behoove the Staff in the draft SEs to identify in each Section, even though they said they can load-follow, we didn't do this.

And the reason why I think it might be

important is, we've had two examples in design certifications where, and I'm sure that won't happen or it may not ever happen with this organization, but other organizations have pulled out midway through and if somebody comes back five years from now and says, oh, we're going to try and certify it again, where are we and why not document it? And why isn't that occurring with the Staff's SEs?

MR. LU: Okay. I think I'll jump and then, Alex, if you want to provide more answer, that would be better. We did see, right at the beginning we saw the DCD document and, oh, we want to have a load-follow. I said, oh, when you have a load-follow, your core design can be quite challenging.

So, the real question here, from 4.3, nuclear design perspective, is, do you really want to do load-follow? When you do load-follow, that means you have very quick core responses, your reactor system, the bank design, the control design, the power shift has to be designed so well that you can really do that a couple minutes load-follow, within very quick transit.

So, we asked them, do you really want to do that? If you want to do that, we are going to impose different requirement on the core side of the design in terms of the reactor near the control, the drive system, and also all the neutronic design perspective. So, we ask the clarification question. They said, oh, every intention was for base load.

MEMBER REMPE: Right.

MR. LU: So, from a 4.3 perspective, because they designed it to go back to the just normal power maneuver, because even you base load a reactor, you start to have startup, your power goes up, right, and then, when the power goes down, you still can go down, it's not something you have constant power all the time throughout the entire cycle.

But they do not do load-follow. So, from the core side, in terms of after they respond to us, that's not their intention, we are fine with the current submitted design, under the 4.3 for the core. Then, let's go back to your question, relate to other Sections, other Chapters --

MEMBER REMPE: And the process.

MR. LU: And the process.

MEMBER REMPE: Why not draw --

MR. LU: Okay.

MEMBER REMPE: Clearly, because I saw this reference to that RAI --

MR. LU: Yes.

MEMBER REMPE: -- where you said, hey, it's just a terminology thing --

MR. LU: Right.

MEMBER REMPE: -- but they have a whole section --

MR. LU: It's not a simple word, from load-follow to power maneuver is quite significant --

MEMBER REMPE: Right.

MR. LU: -- from the core perspective. So, our side is for 4.3 presentation, our SER, is focused on the core side. Now, let's go back to the question related to the turbine. And in turbine, if you have a very quick load drop, the turbine can be fine.

It's not a safety issue from our perspective, from a reactor system design perspective. It's that really issue where you impact the other transient or other in Chapter 15, it should be bounded by the existing Chapter 15 safety analysis.

So, is there really a big problem for other Sections? I think, right now, I think that's a good question and then, I think we resolve the issue for 4.3, the question here is whether it needs to be propagated to other Sections, and that's something we can talk about.

MEMBER REMPE: Well --

MEMBER RAY: Let me make one comment here. The -- you're talking about the core properly and everything you've said, I take no exception to, but the plant can load-follow by bypassing steam to the condenser, just like Palo Verde has 100 percent capability.

MR. LU: True.

MEMBER RAY: So, you've got to be a little careful in talking about the plant, that you're not confusing it with limitations on the core.

MR. LU: That's right. I agree, yes.

MS. KARAS: This is Becky Karas, if I can just add? And I think I understand where you're going philosophically, that if there's in other Sections statements that the plant has the capability to load-follow, right, and if Chapter 4 says something different, I think you're asking, is that a process problem or is that going to cause confusion later when the DC comes for renewal or something like that?

And I think our point is, within Chapter 4, it's clear that that's not the intent and that's not assumed, but in other Sections, if the plant has the capability for load-following, I don't think you would necessarily go and change those Sections if that capability is there.

You still have the restriction on the analytical assumptions and what's okay in terms of Chapter 4, that that would still limit you. And then, you would know what you'd need to look at in the future, if there would want to be a change to that. But it's really a restriction within the Chapter 4.

MEMBER REMPE: But, again, I guess, it's very vague in Chapter 4, you say, hey, they had a terminology issue, we didn't bring it up here, and you refer to that RAI. You don't say, hey, they said they could load-follow and we reviewed it and said they need to change that.

And I just am wondering, is that just the normal way that the Staff would do a design certification and if everyone backed off and said tomorrow, we're done, and that's happened in other design certifications, is it really clearly documented?

Why not say -- and there's been other terminology issues, whether it's a multi-unit or a single-unit plant, and I just am wondering is that normal, and, again, even though I've been on ACRS for quite a few years, the Staff wouldn't clearly say, hey, we didn't agree with them on this and we made them change it? It seems like a pretty significant thing on the load-following.

MS. KARAS: Well, remember, within the design certification, right, the certified design is what's written in the DCD. So, I think I understand, and Alex can correct me if I'm wrong, that the language within Chapter 4 of the DCD, that that language is going to be clarified, I think, as a part of the response, right?

MS. BURJA: That's right.

MS. KARAS: So, that is in fact the design, so with that language being corrected there, then that's what the design is. You don't necessarily have to have an extensive write-up in the SER, because the design itself is what's in the DC.

MS. BURJA: Right.

MS. KARAS: It's not like --

MEMBER REMPE: So, you're telling me,

it's okay that we were kind of vague here and we didn't explicitly call it out, that's just what we've got.

MS. KARAS: Right, you just don't want the words within the DCD that says it's load-following capable within Chapter 4.

MEMBER REMPE: But it does right now in what was submitted.

MS. KARAS: But I think that's what's being --

MEMBER REMPE: The RAI says they're --

MS. KARAS: -- clarified, corrected.

MEMBER REMPE: -- going to change it?

MS. KARAS: Yes.

MEMBER REMPE: Okay.

MS. BURJA: The RAI provided a lot of mark-up, not only of Chapter 4, but of --

MS. KARAS: Right.

MS. BURJA: -- most of the rest of the DCD. And I did a word search in the DCD as well to make sure that it was being changed pretty consistently.

MEMBER REMPE: And you think that you're happy with every change they identified and you think it addresses it?

MS. BURJA: I think so.

MEMBER MARCH-LEUBA: Yes. And from my point of view, which Mike is not here, so I can say, I'm easily confused, he always says that. It's not a semantic issue, but people understand this as semantics, load-follow versus power maneuvers, you're just changing words.

Load-follow is rapid unplanned changes of controlled rods that get through a demand signal. And the key word is unplanned, unanalyzed, you don't know where you're going to end up. Versus power maneuvers are planned control rod movements that they get analyzed before they get exercised.

So, I'm sure at the end of cycle, they'll be running out of reactivity and they will do a cycle stress where they're using the power. That's perfectly okay, that's not load-follow.

So, it would be nice if we defined, let them know, because people get caught up on the terminology instead of what it really means. The thing with the core, what Shanlai, I think, is trying to say, if you're going to accept rapid unplanned rod changes, you need to reanalyze them to make sure that you're okay.

MS. BURJA: Understood. We'll definitely

consider that.

MEMBER MARCH-LEUBA: So, if the SER or the -- could define what they mean, it will go a long time to --

MS. BURJA: Okay. Thank you.

MEMBER MARCH-LEUBA: -- avoiding problems.

MEMBER SUNSERI: Yes. And, Alex, not to belabor the point, but I'll belabor the point. So, I know you said, you just stated you looked through the DCD for changes that affected the reactor, did that include things like, I mean, there's a statement in that says the pressurizer is sized to support load-following and the liquid waste handling systems are sized to support load-following, that all implies the reactor is going to be doing something, right?

MS. BURJA: Right.

MEMBER SUNSERI: So, did your review catch those kind of issues as well?

MS. BURJA: I searched for the term load-follow and some more variations of that. So, it's possible that I missed something, but I can certainly do a more extensive review.

MEMBER SUNSERI: Yes. So, I guess our

caution to the Staff is, this is not just a reactor thing, this is a holistic look at how the DCD is constructed and what's going to ultimately get certified in the end.

MEMBER REMPE: Because the implication is that when they say something like that and you don't say, no, we didn't review this, that -- again, maybe I'm taking it too literally, but someone could come and say, well, we documented it there, the NRC didn't have a problem with it, and that's where I'm coming from. But, again, if that's the way things are done --

MS. KARAS: Yes. This is --

MEMBER REMPE: -- I'm just bringing it up.

MS. KARAS: This is Becky Karas, again. I guess, I would just -- so, it can say in one other Section, and I don't know what all Sections KHNP is modifying, but if one Section said, such-and-such is sized or designed or whatever to support load-following, that doesn't mean that, throughout the DCD, that that's necessarily allowed, right?

So, within Chapter 4, that in and of itself could say, no, power maneuvers, and I think it was characterized as semantics because I think we understood, talking to KHNP, that what they had intended was power maneuvers and not actually to take the plant off of base load. So, you could have certain components that were designed to be able to handled load-following, but that doesn't mean that you're necessarily asking for the design of it to include that capability.

So, as long as within those key Sections, like within the Reactor Section, that it says very clearly that this is base load with power maneuvers, then that's what the design basis of that would be, even if you had design bases of other systems permitting other capabilities.

So, I mean, I think, and we can look at it carefully again and make sure, but my understanding was that we looked and we were relying on KHNP to do that look that they had propagated that throughout their DCD, that the design basis was going to be clear.

MEMBER REMPE: Thanks.

CHAIRMAN BALLINGER: Five seconds. Okay. This is the completion of today's, so far, presentations. I think the next thing we need to do is to solicit public comments, if there are any. And that includes in the room. I don't -- I think the public is not here in the room. But I think we're opening the phone line.

I don't hear any crackling, there's just -- no red light comes on or something. Is there anybody out there on the phone line that would like to make a comment? Is it open? It's open now. There we go.

If there's anybody -- is there anybody out there on the phone line? If you're there, can you make a noise to let us know that you're there? Is there anybody out there that would like to make a comment? Hearing none, we can close the phone line. And next thing is to go around the table and solicit additional comments from members. Joy?

MEMBER REMPE: Thanks to everyone for their work and their presentations today. No additional comments.

MEMBER MARCH-LEUBA: Yes. I also have no further comments. Thank you for your presentations, I think it was excellent presentations, both you and KHNP.

MEMBER KIRCHNER: I thank both parties for their presentations. I just wanted to note, Mr. Chairman, that I felt heartened by the presentations from the Staff that included their own in-house confirmatory analysis of the neutronic design of the core.

And I don't mean this as a plug for research, but I think it's extremely important that the Agency retain that kind of in-house capability to independently validate and verify the designs that they are reviewing. So, I thought that very good.

And just one minor point for Alex, for completeness, I think you should say that the control rod systems can control reactivity changes during DBEs to maintain core cooling and allowable pressure limits on the RCS. That's a factor when you look at RIA, reactivity insertion accidents. Thank you.

CHAIRMAN BALLINGER: Pete?

MEMBER RICCARDELLA: I have no comments.

CHAIRMAN BALLINGER: Matt?

MEMBER SUNSERI: Yes, I'd like to extend my thanks to both the Staff and the Applicant for the thorough discussions that we had today and the patience with our questions. Thank you.

CHAIRMAN BALLINGER: Dana?

MEMBER POWERS: Well, I'm still perplexed a --

CHAIRMAN BALLINGER: Cobalt?

MEMBER POWERS: -- little bit about cobalt, but it seems to me that with all the advances in making hard surfaces, there's a better way to do it, but I don't have to do it, so I'm not going to worry about that.

I am a little bit confused about this compensation for the use of outdated cross-sections by hiding it within biases and offsets and things like that. It is bothersome to me, it seems to me that up-to-date plants ought to use up-to-date databases. That's the only comment I care to make.

CHAIRMAN BALLINGER: Dick?

MEMBER SKILLMAN: Ron, thank you. To both teams, thank you very much. And no further comment.

MEMBER RAY: No comments.

CHAIRMAN BALLINGER: Margaret?

MEMBER CHU: No comments. Thank you.

CHAIRMAN BALLINGER: Steve?

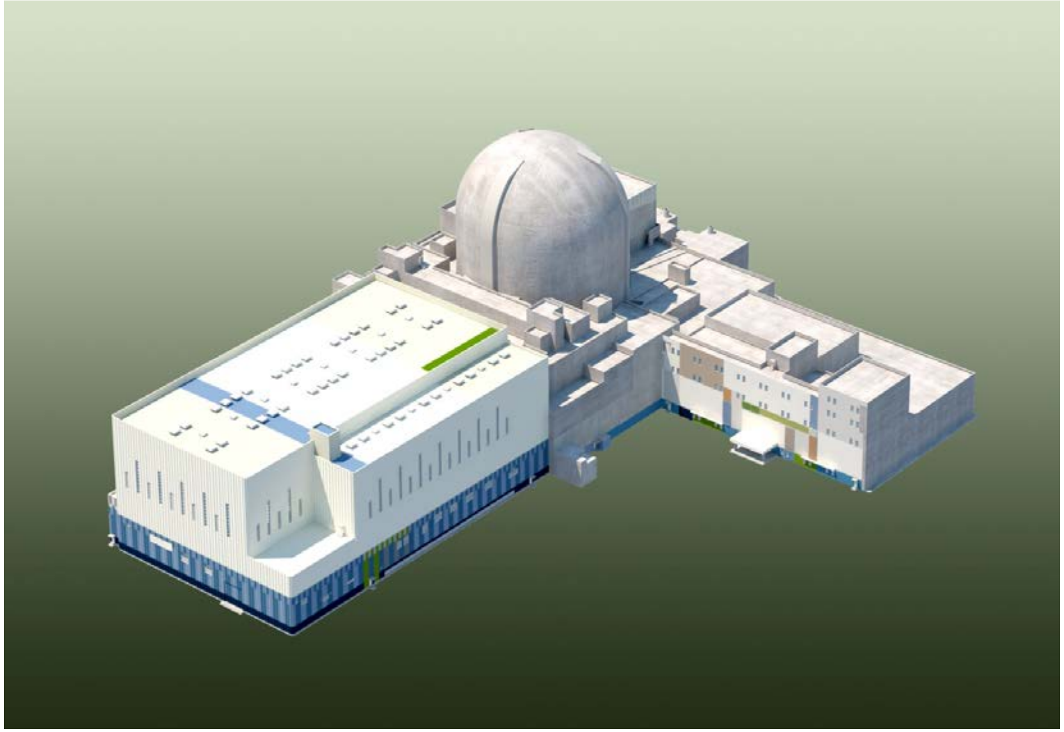
MR. SCHULTZ: I would only second the comments of the Members who have made them here in closing and certainly thank the Staff and the Applicant for good presentations today and a thorough discussion of what has been done so far in this area. It was very well presented, thank you.

CHAIRMAN BALLINGER: And I'd like to extend, second all the other comments, it's been very good today, both on the KHNP side and on the Staff side, and thorough enough so that we can understand what's going on and ask good questions.

And also, I wish we could put the three hours in the bank, but unfortunately we can't, but I can thank you for getting us three hours that we could put in the bank if we could put them in the bank. And other than that, we are adjourned.

(Whereupon, the above-entitled matter went off the record at 1:59 p.m.)

APR1400 DCA
Chapter 4: Reactor



KEPCO/KHNP
Feb. 8. 2017

Contents

Overview of Chapter 4

● Section Overview

Section	Title	Major Contents	Presenter
4.1	Summary Description	• Design features for initial core design and summary information	Ilkyu Kim
4.2	Fuel System Design	• PLUS7 fuel rod and fuel assembly design	Ilkyu Kim
4.3	Nuclear Design	• Nuclear design of APR1400 reactor system	Manseok Do
4.4	Thermal-Hydraulic Design	• Steady-state thermal and hydraulic analysis of the reactor core	Kanghoon Kim
4.5	Reactor Materials	• Materials for CEDM, reactor Internals and core supports	Jongsoo Kim
4.6	Functional Design of Reactivity Control System	• Control Rod Drive System(CRDS)	Jongsoo Kim

NON-PROPRIETARY

Overview of Chapter 4

● References

Submitted Document				Related Section of DCD Chapter 4					
Type	Title	No.	Rev.	4.1	4.2	4.3	4.4	4.5	4.6
DCD	APR1400 Design Control Document Tier 2: Chapter 4 Reactor	APR1400-K-X-FS-14002-NP	0	○	○	○	○	○	○
TOR	PLUS7 Fuel Design for the APR1400	APR1400-F-M-TR-13001-P & NP	0		○		○		
TER	Structural Analysis of Fuel Assemblies for Seismic and Loss of Coolant Accident Loading	APR1400-Z-M-NR-14011-P & NP	0		○				
TER	Neutron Fluence Calculation Methodology for Reactor Vessel	APR1400-Z-A-NR-14015-P & NP	0			○			
TER	Evaluation of Irradiation Assisted Stress Corrosion Cracking and Void Swelling on Reactor Vessel Internals	APR1400-Z-M-NR-14017-P & NP	0					○	
TER	Thermal Design Methodology	APR1400-F-C-NR-12001-P & NP	1				○		
TOR	KCE-1 Critical Heat Flux Correlation for PLUS7 Thermal Design	APR1400-F-C-TR-12002-P & NP	0				○		
TER	CPC Setpoint Analysis Methodology for APR1400	APR1400-F-C-NR-14001-P & NP	0				○		
TER	Functional Design Requirements for a Core Operating Limit Supervisory System for APR1400	APR1400-F-C-NR-14002-P & NP	0			○	○		

* TOR: Topical Report / TER: Technical Report

4.1 Summary Description

- **APR1400 Core and Fuel Design Summary**

Parameters	Values
Core power level (MWt)	3,983
Fuel rod lattice	16 x 16 (236 fuel rods)
Number of fuel assemblies	241
Number of Control Element Assemblies (CEAs)	93
Active fuel length (m)	3.81 (150 in.)
Max. peaking factor (Fq)	2.43
Max. fuel rod avg. burnup (MWD/MTU)	60,000

4.1 Summary Description

● Analytical Techniques

Design Category	Primary Code (Doc. No.)	Analysis Techniques / Approach
Fuel System (Section 4.2)	FATES (CENPD-139-P-A, CEN-161(B)-P-A)	Fuel rod performance analysis
Nuclear (Section 4.3)	DIT (CENPD-266-P-A, CENPD-275-P)	Spectral calculations using discrete integral transport (DIT) theory and spatial calculations in assembly geometry
Nuclear (Section 4.3)	ROCS (CENPD-266-P-A, CENPD-275-P)	Two-group diffusion theory applied with a nodal expansion method (NEM)
Nuclear (Section 4.3)	DORT (Industrial Standard)	Discrete ordinates Sn transport methodology
Thermal-Hydraulic (Section 4.4)	TORC (CENPD-161-P-A)	Subchannel analysis of the local fluid condition in the core

4.1 Summary Description

- **Open Items**
  - **There are no Open Items for Section 4.1.**

4.2 Fuel System Design

- **PLUS7 Design Features**

PLUS7™ FA Design

Fuel Rod	Array	Square, 16 x 16
	No. / FA	236
	Material	UO_2, ZIRLO
Top & Bottom Nozzle	No. / FA	Top: 1, Bottom: 1
	Material	SS304 , Inconel 718
Top & Bottom Grid	No. / FA	Top: 1, Bottom: 1
	Material	Inconel 718
Mid Grid	No. / FA	9
	Material	ZIRLO
Protective Grid	No. / FA	1
	Material	Inconel 718
Guide Thimble	No. / FA	4
	Material	ZIRLO

4.2 Fuel System Design

● PLUS7 Irradiation Experience

- **PLUS7 LTAs (Lead Test Assemblies) and CSAs (Commercial Surveillance Assemblies)**
  - Pool Side Examinations (PSEs) were conducted on four LTAs
  - Hot cell examination of LTA has been completed after irradiation
  - PSEs were conducted on four CSAs
    - ✓ Four CSAs were selected among the fuel assemblies commercially supplied for Hanbit unit 5 cycle 5
  - PSE and hot cell examination results showed all design requirements were met
- **Operating Experience**
  - About 5,000 PLUS7 fuel assemblies have been supplied as of 2016
    - ✓ Shinkori unit 3 started commercial operating with PLUS7 fuel assemblies on Dec. 20, 2016 (Shinkori unit 3 is the reference plant of APR1400)
  - PLUS7 fuel assemblies will be supplied to 4 Barakah APR1400 NPPs in UAE and 5 APR1400 NPPs in KOREA

4.2 Fuel System Design

- **Design Requirements**
  - **10CFR Part 50 Appendix A.**
    - GDC 10 : Reactor design
    - GDC 27 : Combined reactivity control systems capability
    - GDC 35 : Emergency core cooling
  - **10CFR Part 50.46**
- **NRC Guidances**
  - **Regulatory Guide 1.206 Section 4.2**
  - **Standard Review Plan Section 4.2 (NUREG-0800)**
    - Fuel System Damage
    - Fuel Rod Failure
    - Fuel Coolability

4.2 Fuel System Design

● Design Criteria and Evaluation

Evaluation Items	Corresponding Sections
Cladding Stress	4.2.3.1.3, 4.2.3.5.3
Cladding Strain	4.2.3.1.6
Stress and Loading Limit for other than Cladding	4.2.3.4, 4.2.3.5
Cladding Fatigue	4.2.3.1.6
Fretting Wear	4.2.3.1.1, 4.2.3.1.5, 4.2.3.5.2
Cladding Oxidation and Hydriding	4.2.3.1.4
Dimensional Changes	4.2.3.4, 4.2.3.5.2
Rod Internal Pressure	4.2.3.1.2
Assembly Liftoff	4.2.3.5.2
Hydriding	4.2.3.1.4
Cladding Collapse	4.2.3.1.11
Overheating of Fuel Pellets (Melting)	4.2.3.2.3
Pellet-Cladding Interaction (PCI)	4.2.3.3.1

4.2 Fuel System Design

● Open Items

➢ PLUS7 Seismic Technical Report Status

- APR1400-Z-M-NR-14011-P&NP, "Structural Analysis of Fuel Assemblies for Seismic and Loss of Coolant Accident Loading," Rev.0
- Fuel assembly tests and seismic analysis have been performing to answer the RAIs
- RAIs will be responded by Feb. 28, 2017, and the technical report will be revised

RAI NO.	Question No. (Total/Completed/Not responded)
275-8294	8 / 5 / 3
425-8405	6 / 0 / 6

4.3 Nuclear Design

● Nuclear Design Description

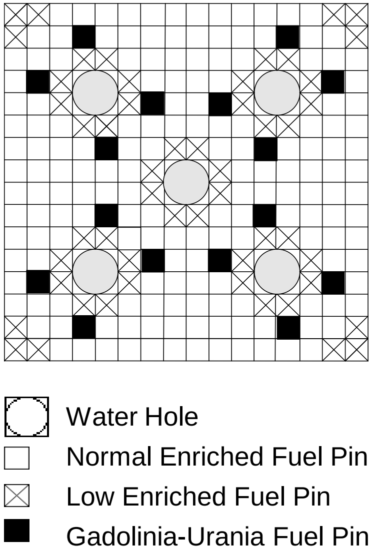
➢ Loading Pattern

- ◆ Three-batch loading scheme with a refueling interval of 18 months

➢ Fuel Assemblies

- ◆ UO_2 fuel rods with enrichment separation and gadolinia-urania (Gd_2O_3-UO_2) burnable absorber rods

Assembly Type	No. of Fuel Assemblies	Fuel Rod Enrichment (w/o)	No. of Gd_2O_3 Rods per Assembly	Gd_2O_3 Contents (w/o)
A	77	1.71	0	-
B	88	3.14/2.64	0, 12, or 16	8
C	76	3.64/3.14	0, 12, or 16	8



Fuel Assembly Batch Information and Typical Configuration for Initial Core

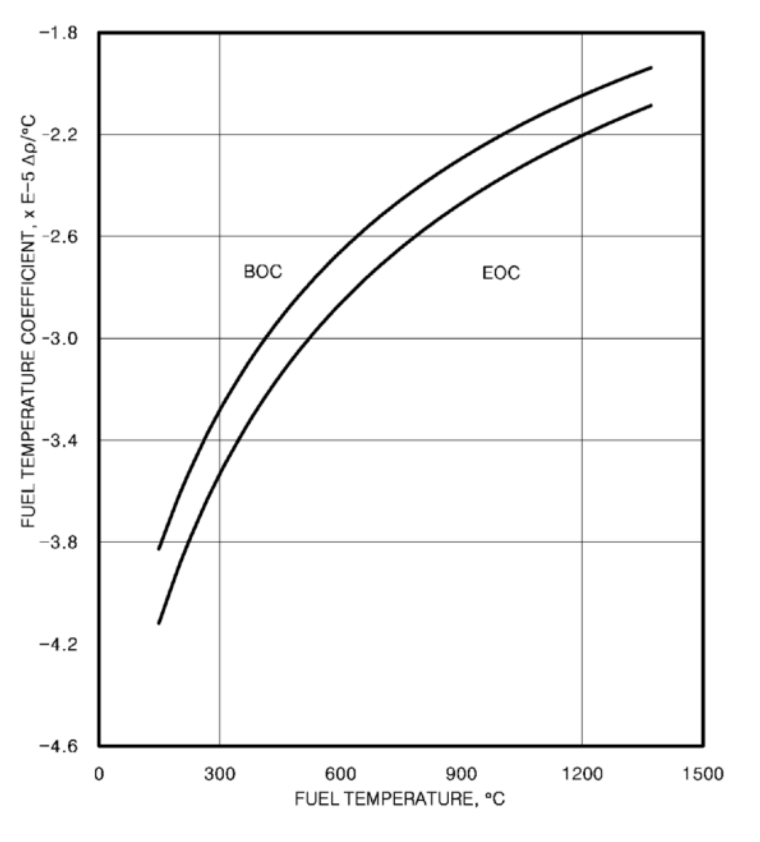
4.3 Nuclear Design

- **Design bases of the APR1400 complies with SRP 4.3**
  - **Power distributions are maintained within the design limits throughout normal operations**
  - **Reactivity coefficients are maintained negative during power operation**
  - **Control systems are capable of providing enough shutdown margin and of controlling power distribution oscillations**

- **Core Power Distributions**
  - **The power distributions during Normal Operation are maintained within the design limits throughout the cycle:**
    - Limiting three-dimensional heat flux peaking factor (F_q) of 2.43
    - Minimum DNBR of 1.29
    - Maximum peak fuel rod burnup of 60 GWD/MTU
  - **The design limits on the power distribution are used both as design input and as initial conditions for accident analyses**

4.3 Nuclear Design

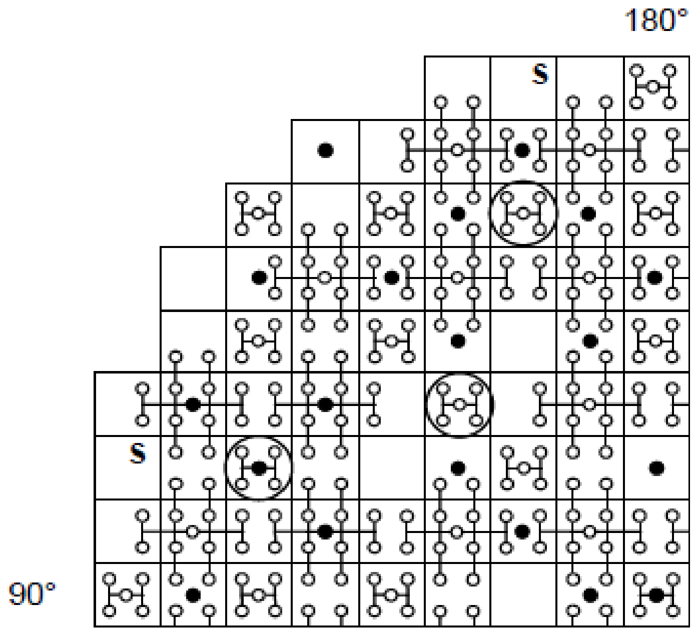
- **Reactivity Coefficients**
  - **Fuel Doppler Temperature Coefficient**
    - Negative throughout the cycle
  - **Moderator Temperature Coefficient**
    - Negative for most of power operation ranges
    - Burnable absorbers achieve a negative coefficient at BOC



4.3 Nuclear Design

● Reactivity Control System

- Provides enough shutdown margin considering single malfunctions of the reactivity control systems
- B_4C for Full-strength CEA and Inconel for Part-strength CEA
- Power dependent insertion limits (PDILs) conform to shutdown margin



CEA Groups	No. of CEAs
Regulating CEA Group	45
Shutdown CEA Group	36
Part Strength CEA Group	12
Total	**93**

4.3 Nuclear Design

- **Stability**
  - **Total core power perturbation**
    - Inherently stable because of the negative overall power coefficients
  - **Power distribution perturbation**
    - Stable for radial and azimuthal xenon-induced oscillations
    - Monitoring and protection by COLSS and CPCS
    - Effective control by PSCEAs or regulating CEAs

4.3 Nuclear Design

- **Open Items**
  - **There are no Open Items for Section 4.3.**

4.4 Thermal-Hydraulic Design

● Design Bases

Parameters	Bases	Values
DNBR Limit	95% Probability/95% Confidence	1.29
UO_2 Max. Temperature	Under melting temperature	2,804 ^oC (5,080 ^oF) *
RCS Flow Rate	Greater than minimum / Less than maximum	Min.:100%Q_D Max.:115%Q_D
Hydraulic Instability	Not occur	←

● Thermal-Hydraulic Major Parameters

Parameters	Values
Total core heat output	3,983 *MWt*
Primary system pressure	158.2 kg/cm^2 (2,250 *psia*)
Reactor inlet coolant temperature	290.6 ^oC (555 ^oF)
Design primary coolant flow rate(Q_D)	1,689,000 *L/min* (446,300 *gpm*)
Minimum DNBR at nominal condition	2.44
UO_2 Max. Temperature at nominal condition	1,712 ^oC (3,114 ^oF)

* @ BOC

4.4 Thermal-Hydraulic Design

- **Reactor Vessel Flow Distribution**

  - **Main flow path**

    Inlet Nozzle → Downcomer

    → Flow Skirt → Lower Plenum

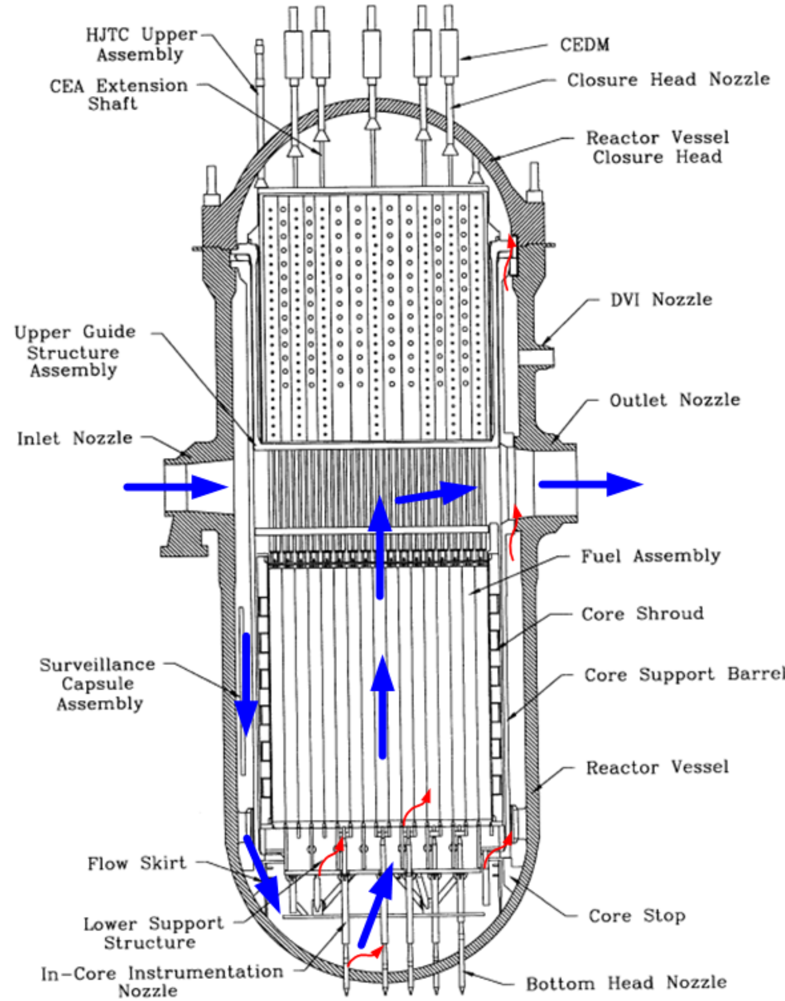
    → Core → Fuel Alignment Plate

    → Upper Plenum → Outlet Nozzle

  - **Core bypass flow**

    3% of total vessel flow

  - **Core flow rate is determined to ensure an adequate core cooling**



4.4 Thermal-Hydraulic Design

- **Thermal Effects of Operational Transients**
  - Design basis limits on DNBR and fuel temperature are maintained by LCO in the Technical Specifications for the most limiting AOO
- **Uncertainties for DNBR Calculation**
  - Input to build the core analysis model
  - TORC analytical model
  - CHF (DNB) correlation
- **Critical Heat Flux**
  - KCE-1 CHF correlation was used with TORC and CETOP codes to calculate DNBR for normal operation and AOOs
  - This correlation was developed based on PLUS7 CHF test data
  - Topical Report of KCE-1 CHF correlation (APR1400-F-C-TR-12002) → ACRS reviewed on Dec. 14, 2016

4.4 Thermal-Hydraulic Design

- **Core Thermal Response**
  - **COLSS and RPS provide reasonable assurance that the design bases are not violated for any normal operating condition and AOOs**
- **Analytical Methods**
  - **Reactor coolant system flow was determined by the system flow resistance and the RCP performance**
  - **Thermal Margin Analysis were performed by TORC/CETOP codes and SCU method, previously approved by NRC**

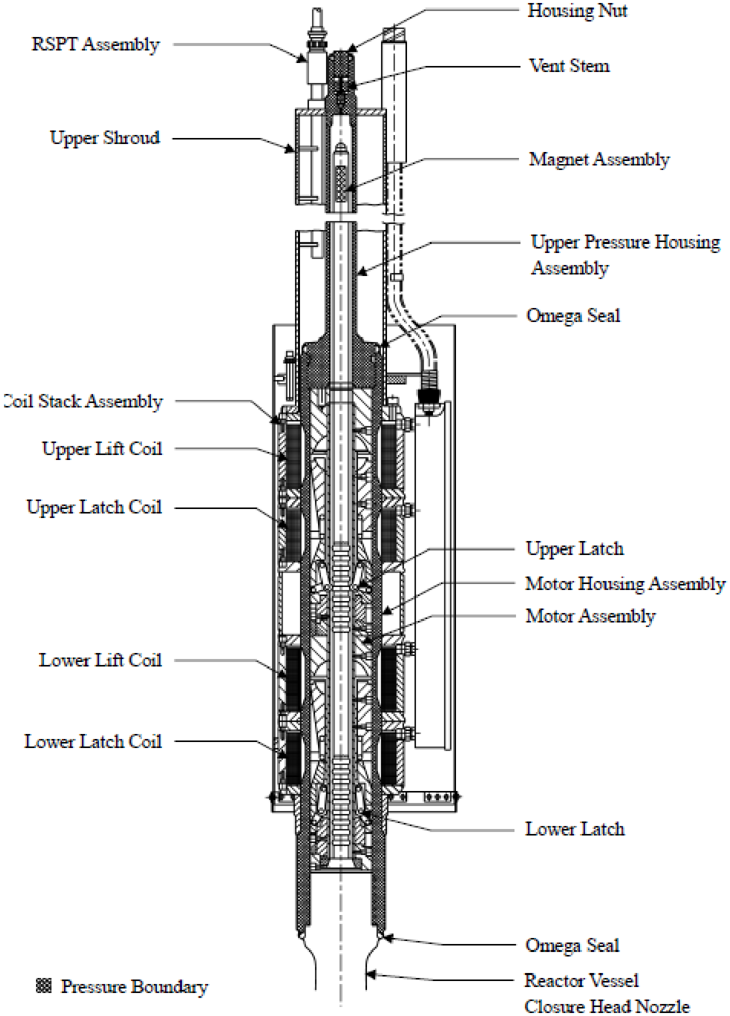
4.4 Thermal-Hydraulic Design

● Open Items

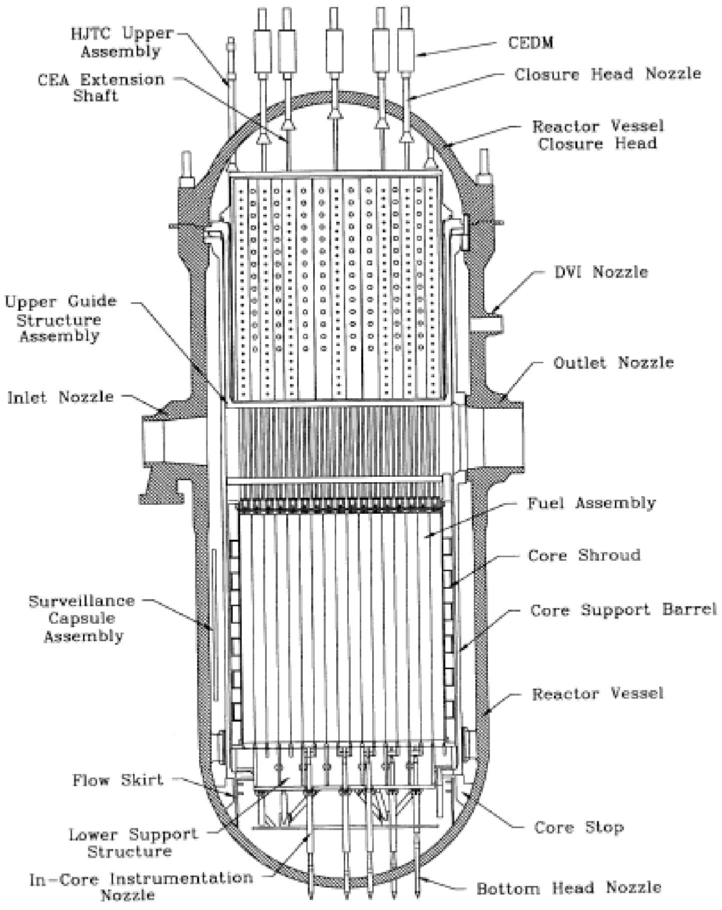
RAI No.	Question No.	RAI Topic / NRC Concern	RAI Response / DCD Impact
328-8422	04.04-7	• Addition of APR1400-F-C-NR-14002-P to Reference in the DCD Tier 2	(1) RAI response was submitted on 8/19, 2016. (ML16232A569) (2) Impact on DCD (Revision of DCD Tier 2, Table 1.6-2)
328-8422	04.04-8	• Addition of the methodology for mixed cores to the DCD Tier 2 or corresponding TER	(1) RAI response was submitted on 8/29, 2016. (ML16242A432) (2) Impact on DCD (Revision of DCD Tier 2, Table 1.6-2, TER : APR1400-F-C-NR-14001-P)

4.5 Reactor Materials

- **Control Element (Rod) Drive Mechanism**
  - Function : to control reactivity (DCD Section 3.9.4)



- **Reactor Vessel Internals (RVI)**
  - Function : to support fuel and maintain flow in RV (DCD Section 3.9.5)



4.5 Reactor Materials

- **CEDM Materials –Material Specifications**
  - **RCPB materials in CEDM**
    - Motor Housing Assembly
      - ✓ Martensitic stainless steel, Austenitic stainless steels, Nickel-base alloys
    - Upper Pressure Housing Assembly
      - ✓ Austenitic stainless steels
    - The materials used in RCPB comply with the requirements of ASME Sections III, II, IX and Reg. Guide 1.84.
  - **Reactor coolant contact materials in CEDM**
    - Internal components of CEDM (Motor Assembly, Extension Shaft Assembly)
    - Corrosion resistant materials; Austenitic stainless steels, Martensitic stainless steels, Nickel base alloys, Cobalt alloys

4.5 Reactor Materials

- **CEDM Materials – Material Specifications**

  - **Weld materials in CEDM**
    - Austenitic stainless steel weld metals (SFA-5.9 ER 316L)
    - Nickel-based weld metals (SFA-5.14 ERNiCrFe-7A)

  - **The materials used in CEDM Assembly of APR1400 are:**
    - Essentially identical to Palo Verde and 12 operating OPR1000 plants in Korea, which demonstrate that the CEDM operates without malfunction with these materials.
    - Tested to exceed the life-time requirements (DCD Section 3.9.4)

4.5 Reactor Materials

- **CEDM Materials**
  - **Austenitic Stainless Steel Components**
    - Only proved procedures will be applied to CEDM fabrication
    - Process controls for RCPB (DCD Sec. 5.2.3.4) are applied on CEDM
    - Venting of CEMD will be applied on each refueling period before plant starts
  - **Other Materials**
    - Nickel base alloys, Cobalt alloys, and Martensitic stainless steels for Springs, Grippers, Latches and Links, Bearing Inserts, Alignment Tabs, and Steel Balls

4.5 Reactor Materials

- **RVI Materials – Materials Specifications**
  - **Reactor internals and core support (RVI) materials**
    - Comply with ASME Section III NG-2000 and Reg. Guide 1.84
    - Primarily Type 304 austenitic stainless steels
    - Welded connections are applied where feasible, structural fasteners are typically Type 316
    - Cobalt hardfacing materials at wear points
    - Cold worked austenitic stainless steel will not be used except for bolting or pins
    - Proven materials which performed satisfactorily in operating reactors in US and Korea
  - **Reactor internals and core support materials specifications**
    - Core support barrel assembly, Upper Guide Assembly, Core Shroud Assembly
      - ✓ Mainly austenitic stainless steels

4.5 Reactor Materials

- **RVI Materials –Materials Specifications**
  - **Reactor internals and core support materials specifications**
    - Bolt and pin material
      - ✓ Austenitic Stainless Steels
    - Chrome plating and Cobalt hardfacing
    - Weld materials for RVIs
      - ✓ Stainless steel weld metal of 308L (SFA-5.4, SFA-5.9), Stellite 25 for hardfacing weld
  - **Controls on Welding and Nondestructive Examination**
    - Welding and Examination of Internals and Core support materials comply with ASME Section III (NG) Code
  - **Fabrication and Processing of Austenitic Stainless Steel**
    - Reg. Guide 1.44 are applied to control the use of sensitized austenitic stainless steel
    - Process controls for RCPB (DCD Sec. 5.2.3.4) applied on RVI austenitic stainless steels

4.5 Reactor Materials

- **Reactor Internals and Core Support Materials**

  - **Other Materials**
    - Austenitic stainless steels (Grade 660, S21800) and Martensitic stainless steels (F6NM) for Alignment Keys, Insert pins, HJTC tube assembly and Hold down ring

  - **Other Degradation Mechanisms (Irradiation Assisted Stress Corrosion Cracking (IASCC) and Void Swelling)**
    - IASCC and Void Swelling become challenging degradations for RVI materials
    - Neutron fluence, temperature and stresses are main influencing factors
    - EPRI developed software applied for the assessment for APR1400 RVI
    - The assessment results of APR1400 RVIs is acceptable
    - Evaluation report was provided to NRC

4.5 Reactor Materials

- **Open Items**

RAI No.	Question No.	RAI Topic / NRC Concern	RAI Response
523-8684	04.05-01-15	DCD Section 4.5.1.1 be revised to include the material specifidcaitons and types for the Versa Vent™ since it is a pressure boundary component	RAI response was submitted on 10/25, 2016.
523-8684	04.05-01-16	Data/operational experience that demonstrates the venting with Versa Vent™ can work in practice to eliminate the air trapped in the top of the CEDM	RAI response was submitted on 11/15, 2016.

4.6 Functional Design of Reactivity Control System

- **The section 4.6 describes the Control Rod Drive System (CRDS) that consists of the CEDMs, which insert or withdraw the CEAs, and the Digital Rod Control System (DRCS), which actuates the CEDMs.**

- **Information and Evaluation of combined performance of the reactivity control systems are about the design bases events analyzed in chapter 15 that require reactivity control systems to operate for preventing or mitigating each event.**

4.6 Functional Design of Reactivity Control System

- **Open Items**
  - **There are no Open Items for Section 4.6.**

Summary

- **APR1400 reactor design of Chapter 4 demonstrates to comply with requirements of federal regulations and NRC regulatory documents.**
- **There are no Open Items for Sections 4.1, 4.3 and 4.6.**
- **There are 6 Open Items in total for Sections 4.2, 4.4 and 4.5.**

Attachment: Acronyms (1/3)

- ACRS : Advisory Committee on Reactor Safeguards
- AOO: Anticipated Operational Occurrence
- APR1400: Advanced Power Reactor 1400
- ARO: All Rods Out
- ASME: American Society of Mechanical Engineers
- BOC: Beginning of Cycle
- CEA: Control Element Assembly
- CEDM: Control Element Drive Mechanism
- CFR: Code of Federal Regulations
- CHF: Critical Heat Flux (or DNB : Departure from Nucleate Boiling)
- COLSS: Core Operating Limit Supervisory System
- CPCS: Core Protection Calculator System
- CRDM: Control Rod Drive Mechanism
- CSAs: Commercial Surveillance Assemblies

Attachment: Acronyms (2/3)

- DNBR: Departure from Nucleate Boiling Ratio
- EOC: End of Cycle
- FTC : Fuel Temperature Coefficient
- HJTC: Heated Junction Thermocouple
- IASCC: Irradiated Assisted Stress Corrosion Cracking
- GDC: General Design Criteria
- KHNP: Korea Hydro & Nuclear Power
- LCO: Limiting Conditions for Operation
- LTAs: Lead Test Assemblies
- MTC: Moderator Temperature Coefficient
- PCI: Pellet-Cladding Interaction
- PDIL: Power Dependent Insertion Limit
- PSCEA: Part Strength Control Element Assembly
- PSE: Pool Side Examination

Attachment: Acronyms (3/3)

- RAI: Request for Additional Information
- RCP : Reactor Coolant Pump
- RCPB: Reactor Coolant Pressure Boundary
- RCS: Reactor Coolant System
- RG: Regulatory Guide
- RPS : Reactor Protection System
- RVI: Reactor Vessel Internals and Core Support

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Protecting People and the Environment

Presentation to the ACRS Subcommittee

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

Safety Evaluation with Open Items: Chapter 4

REACTOR

February 8, 2017

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Technical Topics
Section 4.2 – Fuel System Design

Areas of Review

- Fuel System Design
- Nuclear Design
- Thermal and Hydraulic Design
- Materials
- Reactivity Control

Technical Topics
Section 4.2 – Fuel System Design

Areas of Review

- Design Bases
  - Fuel assembly damage
  - Fuel rod damage
  - Core coolability
- Descriptions and Design Drawings
- Design Evaluation
- Testing, Inspection, and Surveillance Plans
- ITAACs
- COL Action Items and Certification Restrictions

Technical Topics

Section 4.2 – Fuel System Design

Challenging Review Area: Burnup Dependent Thermal Conductivity Degradation

- APR1400 Fuel System Design safety analysis is based on the use of the FATES-3B fuel design code, which does not contain a burnup dependent thermal conductivity degradation (TCD) model
- The staff identified concerns regarding compliance with GDC 10 for various fuel system damage and fuel rod damage mechanisms as well as 10 CFR 50.46 for core coolability requirements when the burnup dependency of TCD is not modeled.
- The burnup dependent TCD model review area is addressed as part of the ongoing reviews of the referenced topical reports APR1400-F-M-TR-13001-P "PLUS7 Fuel Design for the APR1400" and APR1400-F-A-TR-12004-P, "Realistic Evaluation Methodology for Large-Break LOCA of the APR1400". The resolution of the DCD Section 4.2 open item depends on the successful completion of these associated topical reports.

<u>Challenging Review Area: Fuel Assembly Structural Response to Externally Applied Loads</u>

- During review of the fuel assembly structural response analysis, the staff noted that the referenced methodology was not strictly followed in its entirety calling into question the determination of load limits for the PLUS7 fuel assembly.
- The applicant is in the process of completing its open item resolution plan which includes a complete test program of the PLUS7 fuel assembly and grid for both beginning of life (BOL) and end of life (EOL) conditions.
- The staff has been auditing the tests as they occur and will be able to review the final analysis documentation when the open item resolution plan has been completed.

Technical Topics

Section 4.2 – Fuel System Design

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Findings – Fuel System Design meets the following requirements:

- The staff is currently unable to make regulatory findings on the fuel system design criteria due to open items associated with the ongoing topical report reviews of APR1400-F-M-TR-13001-P "PLUS7 Fuel Design for the APR1400" and APR1400-F-A-TR-12004-P, "Realistic Evaluation Methodology for Large-Break LOCA of the APR1400", as well as the open item resolution plan associated with technical report APR1400-Z-M-NR-14010-P, "Structural Analysis of Fuel Assemblies for Seismic and Loss of Coolant Accident Loading".

Technical Topics
Section 4.3 – Nuclear Design

Areas of Review

- Design Bases
- Power Distributions
  - Representative power distributions
  - Monitoring: core operating limits supervisory system (COLSS) and core protection calculator system (CPCS)
  - Means of control: core loading pattern, control element assemblies (CEAs), soluble boron
- Reactivity Coefficients
  - Negative power coefficient of reactivity
  - Conservative values used in transient and accident analyses
- Reactivity Control Provisions and Requirements
  - Available CEA worth sufficient for safe shutdown during normal and accident conditions
  - Conservative worth values used in transient and accident analyses
  - Reactivity control requirements are clearly defined and reasonable
  - Limits on reactivity insertion rate
- Provisions to prevent reactor criticality during refueling

Technical Topics
Section 4.3 – Nuclear Design

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Areas of Review, Continued

- Stability against xenon-induced power distribution oscillations
- Analytical Methods
  - Nuclear design methodology - DIT and ROCS
  - Codes used to process information from the ex-core detectors for use in the CPCS
- Reactor Vessel Fluence
  - Methodology and assumptions - DORT, a 2-D discrete ordinates transport code
  - Vessel fluence calculation, including associated bias and uncertainty
  - Combined license (COL) information item for plant-specific surveillance data for benchmarking (COL Information Item 5.3(5))
- Initial test program – nuclear design tests
  - Open Item 14.2.12.2-1 related to the absence of initial fuel load and initial criticality tests now a Confirmatory Item; response to RAI acceptable
- TS: safety limits and TS related to power distribution, reactivity control, and instrumentation

Technical Topics
Section 4.3 – Nuclear Design

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Challenging Review Area: Control Rod Worth Depletion

- Staff concern: Full-strength CEAs with B_4C neutron absorber may be used as regulating rods, and B-10 depletion may affect CEA worth
- Resolution
  - Applicant's estimated B-10 burnout not negligible but conservative
  - Operating experience for a similar reactor design (Optimized Power Reactor 1000 (OPR1000)) shows measurements agree with predictions within allowed uncertainty
  - Net rod worth uncertainty listed in the DCD much greater than estimated loss of worth
  - Shutdown reactivity curve is already conservative
  - Ten-year CEA lifetime and power-dependent insertion limits limit loss of worth
  - Startup physics tests confirm rod worth is consistent with predictions

Technical Topics
Section 4.3 – Nuclear Design

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Challenging Review Area: Nuclear Design Methodology – Benchmarking

- Staff concern: DCD did not adequately describe benchmarking of DIT and ROCS against experimental data for the APR1400-specific nuclear and fuel design
- Resolution
  - Bias and uncertainty manual derived from measured vs. predicted data for 8 US Combustion Engineering reactors with core and fuel designs that bound the APR1400 design
  - Audit of calculation notes confirmed methods to determine bias and bias uncertainties
  - Audit of bias and uncertainty manual confirmed the high-level information in the DCD
  - APR1400 bias and uncertainty manual almost identical to that for OPR1000
  - Staff confirmed that the OPR1000 DIT and ROCS predictions provided in RAI response compare well against plant measurements

Challenging Review Area: Nuclear Design Methodology – Nuclear Data

- Staff concern:
  - Evaluated Nuclear Data File (ENDF)/B-IV cross-section library published in 1974; many important improvements since then
  - Staff performed confirmatory criticality calculations using the SCALE code to compare the results when using the ENDF/B-IV library instead of the current version; differences substantial (see next slide)
  - Use of ENDF/B-IV could lead to inaccurate nuclear design predictions and affect transient and accident analyses
- Resolution
  - Bias and uncertainty values applied to DIT and ROCS are associated with use of the ENDF/B-IV library; thus, effects of library implicitly captured in the bias and uncertainty applied to the nuclear design calculations

Technical Topics
Section 4.3 – Nuclear Design

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Challenging Review Area: Nuclear Design Methodology – Nuclear Data

- Staff confirmatory calculation results

LEU-COMP-THERM-001, Case 1			
Cross-Section Library	k_{eff}	σ	Difference compared to ENDF/B-VII (pcm)
ENDF/B-VII (Continuous Energy)	0.99885	0.00099	0.00
ENDF/B-V (238-Group)	0.99584	0.00091	-301.35
ENDF/B-IV (218-Group)	0.99061	0.00088	-824.95

APR1400 3.14 wt% UO_2 Unpoisoned Fuel Assembly			
Cross-Section Library	k_∞	σ	Difference compared to ENDF/B-VII (pcm)
ENDF/B-VII (Continuous Energy)	1.40879	0.00044	0.00
ENDF/B-V (238-Group)	1.40049	0.00043	-589.16
ENDF/B-IV (218-Group)	1.38736	0.00044	-1521.2

Technical Topics

Section 4.3 – Nuclear Design

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Conclusions

- Meets GDC 10, 11, 12, 13, 20, 25, 26, 27, and 28 because:
  - Specified acceptable fuel design limits (SAFDLs) will not be exceeded during normal operation or anticipated operational occurrences (AOOs)
  - Negative power coefficient of reactivity
  - Only axial xenon oscillations are possible; can be suppressed using CEAs
  - Instrumentation and controls to maintain power distributions within limits
  - Control systems and setpoints for shutdown adequate
  - SAFDLs not exceeded for any single reactivity control system malfunction
  - CEAs and chemical and volume control system (CVCS) provided
  - Sufficient shutdown margin assuming a stuck rod
  - Reactivity insertion limited, appropriate reactivity values in transient and accident analyses
- Additional Considerations
  - Analytical methods and data acceptable, and methods benchmarked appropriately
  - Initial test program adequate to verify the nuclear design
  - Nuclear design TS ensure protection of public health and safety
  - Staff confirmatory analyses support the foregoing conclusions

Staff Initial Cycle Confirmatory Analyses: Outline

- Description of NRC nuclear design methods
- Comparison of power distributions
- Comparison of boron letdown
- Comparison of control bank worths
- Comparison of trip reactivity insertion curves
- Reactivity Feedback Coefficients
- Summary and Conclusions

Technical Topics
Section 4.3 – Nuclear Design
Staff Initial Cycle Confirmatory Analyses

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NRC Methods

- Lattice Physics calculations are performed with SCALE/POLARIS
  - 7 history conditions
  - 81 branch conditions
  - Covers normal operation and anticipated operational occurrences
- Nodal core simulator neutronic calculations performed with PARCS
  - 27 total nodes, 25 active nodes axially
  - 1 radial node per fuel assembly
  - 250 MWd/MTU cycle exposure increments
- Nodal core simulator thermal-hydraulic calculations performed with PATHS
  - 1 thermal-hydraulic channel per fuel assembly

Technical Topics
Section 4.3 – Nuclear Design
Staff Initial Cycle Confirmatory Analyses

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Radial Power Distribution – Beginning of Cycle (BOC)

					0.72	0.94	1.06	1.02
			0.78	1.09	1.00	1.08	1.05	1.12
		0.82	1.08	1.14	0.97	1.13	0.96	1.05
	0.78	1.08	1.03	0.95	1.05	0.96	1.16	0.97
	1.09	1.14	0.95	1.14	0.92	1.11	0.95	1.14
0.72	1.00	0.97	1.05	0.92	1.02	0.90	1.01	0.91
0.94	1.08	1.13	0.96	1.11	0.90	1.09	0.88	1.05
1.06	1.05	0.96	1.16	0.95	1.01	0.88	0.97	0.88
1.02	1.12	1.05	0.97	1.14	0.91	1.05	0.88	0.92

DCD Reference

					0.69	0.86	0.95	0.92
			0.75	1.04	0.96	1.02	0.99	1.06
		0.80	1.05	1.10	0.96	1.11	0.96	1.04
	0.75	1.05	1.02	0.96	1.06	0.98	1.17	1.00
	1.04	1.10	0.96	1.15	0.97	1.15	1.00	1.18
0.69	0.96	0.96	1.06	0.97	1.07	0.97	1.07	0.98
0.86	1.02	1.11	0.98	1.15	0.97	1.15	0.96	1.13
0.95	0.99	0.96	1.17	1.00	1.07	0.96	1.05	0.96
0.92	1.06	1.04	1.00	1.18	0.98	1.13	0.96	1.00

PARCS Result

Technical Topics
Section 4.3 – Nuclear Design
Staff Initial Cycle Confirmatory Analyses

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Radial Power Distribution – End of Cycle (EOC)

					0.61	0.76	0.83	0.81
			0.65	0.88	0.98	1.09	1.12	1.19
		0.74	1.08	1.10	0.95	1.20	0.97	1.15
	0.65	1.08	1.15	0.97	1.14	0.96	1.12	0.94
	0.88	1.10	0.97	1.22	0.97	1.19	0.95	1.11
0.61	0.98	0.95	1.14	0.97	1.15	0.97	1.13	0.95
0.76	1.09	1.20	0.96	1.19	0.97	1.21	0.96	1.19
0.83	1.12	0.97	1.12	0.95	1.13	0.96	1.13	0.93
0.81	1.19	1.15	0.94	1.11	0.95	1.19	0.93	0.90

DCD Reference

					0.65	0.78	0.86	0.83
			0.68	0.91	1.00	1.10	1.14	1.21
		0.77	1.09	1.11	0.93	1.21	0.96	1.17
	0.68	1.09	1.15	0.95	1.15	0.94	1.13	0.93
	0.91	1.11	0.95	1.22	0.94	1.18	0.92	1.11
0.65	1.00	0.93	1.15	0.94	1.14	0.93	1.12	0.92
0.78	1.10	1.21	0.94	1.18	0.93	1.19	0.92	1.16
0.86	1.14	0.96	1.13	0.92	1.12	0.92	1.10	0.89
0.83	1.21	1.17	0.93	1.11	0.92	1.16	0.89	0.87

PARCS Result

Technical Topics
Section 4.3 – Nuclear Design
Staff Initial Cycle Confirmatory Analyses

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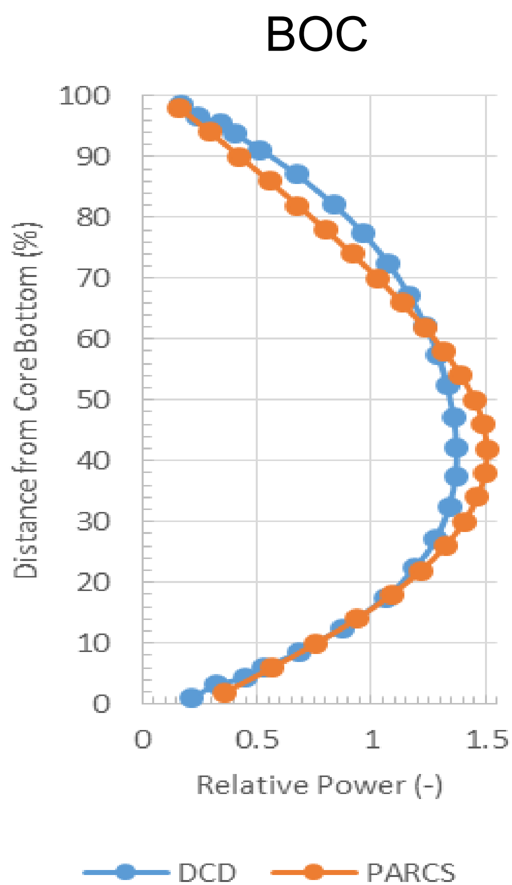
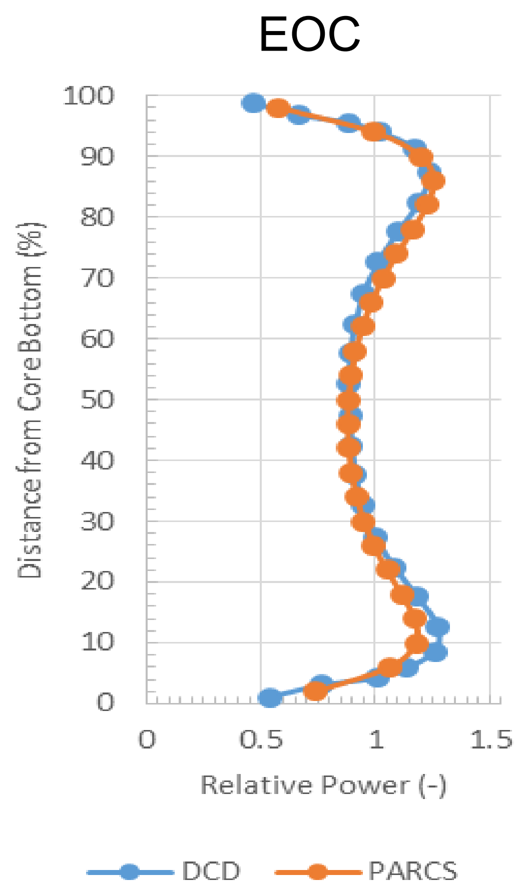
Axial Power Distribution – Beginning and End of Cycle





Technical Topics
Section 4.3 – Nuclear Design
Staff Initial Cycle Confirmatory Analyses

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Power Distribution Results

- The differences in radial power predictions were found to be reasonable and explainable, and there were minimal differences between the staff calculation and the applicant prediction of the highest-powered assemblies.
- The differences in axial peaking between the two cases are, at most, ~10% at certain points in cycle. This is consistent with PARCS assessment against plant data.
- At no time throughout the initial cycle are the results close to the axial offset limits set by APR1400 technical specifications.

Technical Topics
Section 4.3 – Nuclear Design
Staff Initial Cycle Confirmatory Analyses

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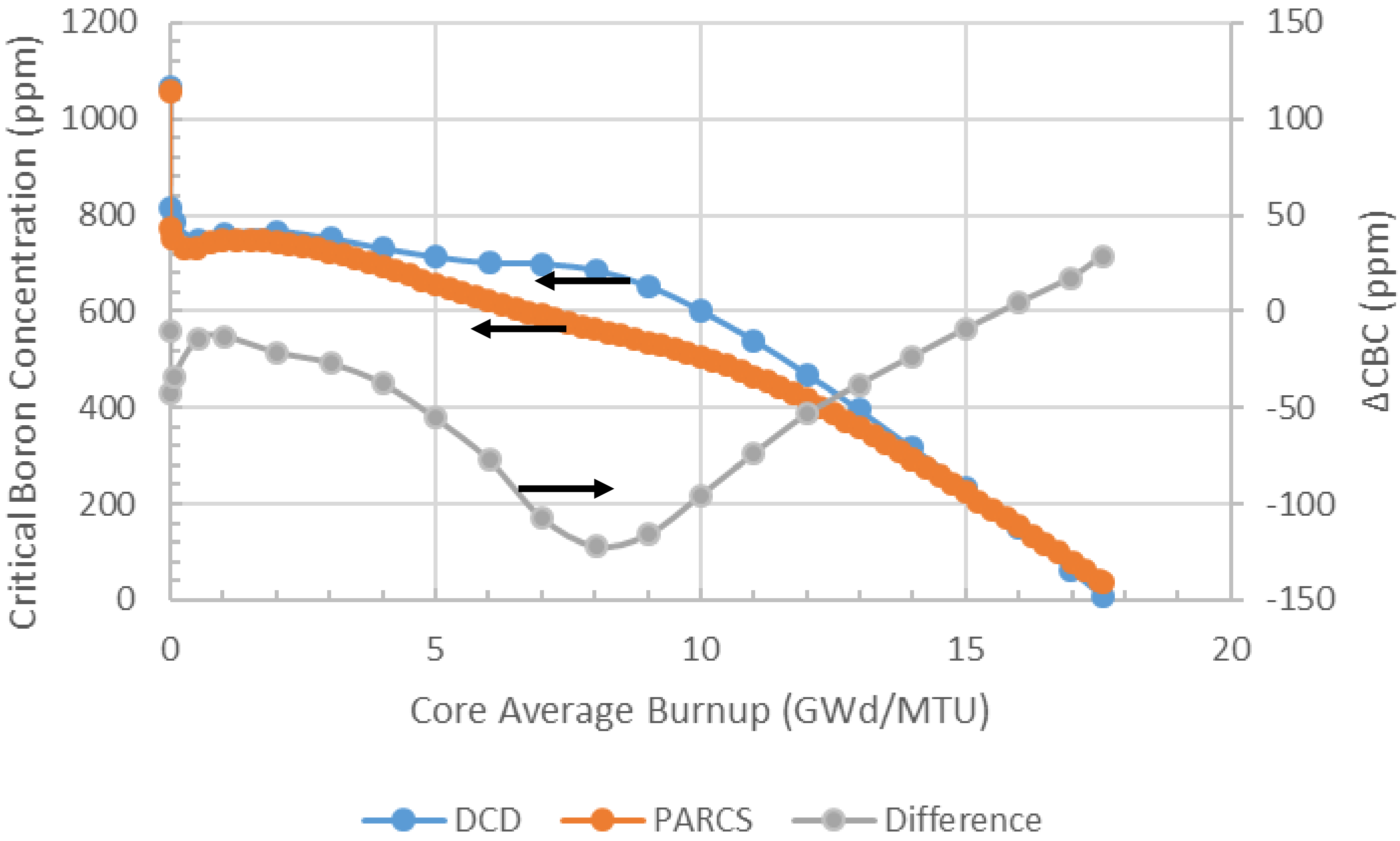
Boron Letdown Comparison



Technical Topics
Section 4.3 – Nuclear Design
Staff Initial Cycle Confirmatory Analyses

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Boron Letdown Conclusions

- The qualitative behavior of the NRC and applicant calculations is very similar.
- Starting from ~5 GWd/MTU of burnup through ~12 GWd/MTU, the deviation in critical boron concentration exceeds 50 ppm, with the applicant's calculations requiring a higher critical boron concentration. The differences in critical boron concentration are likely caused by differences in isotopics or fuel/moderator conditions.
- Mid-cycle conditions are not limiting in safety analysis, and at the limiting (i.e., BOC and EOC) points, the PARCS and applicant's analysis agreement is reasonable-to-excellent. Therefore, while the difference between DCD and PARCS predictions may be worthy of further study, they are not significant enough to warrant safety concern.

Technical Topics

Section 4.3 – Nuclear Design

Staff Initial Cycle Confirmatory Analyses

Control Rod Worth Comparison

	Difference in Worth (%)			
Bank	**0 MWd/MTU**	**7018 MWd/MTU**	**13992 MWd/MTU**	**17571 MWd/MTU**
1	13.60	1.17	2.03	1.25
2	-3.87	-4.24	-8.80	-10.42
3	-0.34	-6.59	-8.62	-10.20
4	-1.74	1.87	-2.18	-2.10
5	0.60	-6.41	0.00	0.25
SD	-2.22	-4.78	-4.76	-6.63
PS	-5.09	-9.23	-9.33	-8.24
Total	-0.58	-3.84	-4.07	-5.68

Generally excellent agreement between the DCD and the confirmatory analysis is evidenced. As a point of reference, a bank worth difference between expected and actual of approximately 10% is considered satisfactory for most core design applications.

Technical Topics
Section 4.3 – Nuclear Design
Staff Initial Cycle Confirmatory Analyses

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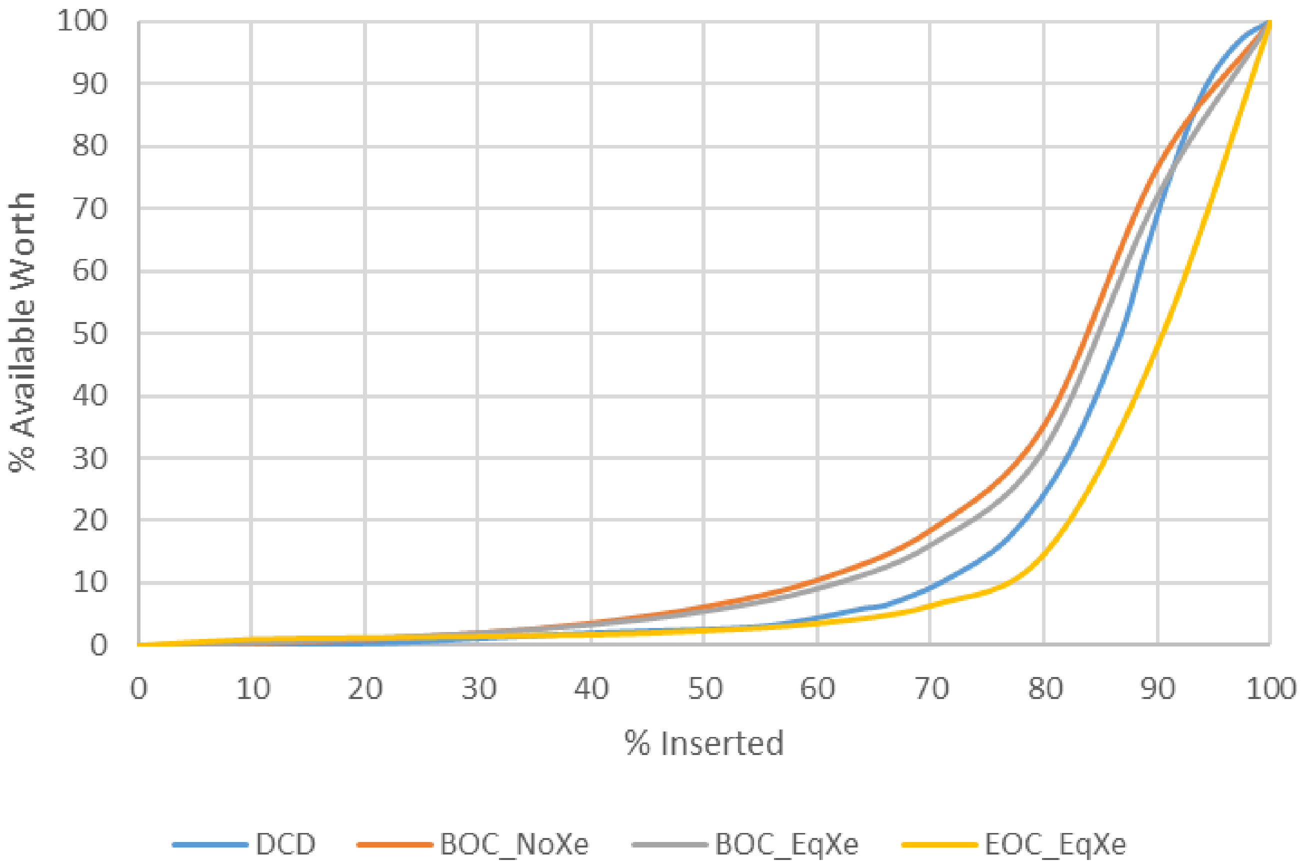
Reactor Trip Reactivity Insertion



Technical Topics
Section 4.3 – Nuclear Design
Staff Initial Cycle Confirmatory Analyses

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Reactivity Feedback Coefficients

- Moderator Density Coefficient (MDC) and Fuel Temperature Coefficient (FTC) were calculated at BOC and EOC conditions. Limiting point is BOC without Xenon/Samarium.
  - MDC is ~ 9.7×10^{-3} \$-m^3/kg
  - FTC is ~ -2.1×10^{-3} \$/K
- Power feedback coefficient confirmed to be negative at power conditions.
  - Slightly negative MDC at very low power, fresh core. This is counter balanced by negative FTC at these conditions.
- Data can be used in conjunction with TRACE to perform transient calculations using point kinetics.

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Summary and Conclusions

- Radial power distribution had a difference of ~2%-5% depending on time-in-cycle. Comparisons in the peak powered assemblies were on the order of 1%. This agreement is reasonable-to-excellent.
- There is excellent agreement in axial power distribution with differences less than ~10%.
- The boron letdown curve shows excellent qualitative agreement, but in the middle of the cycle differences are beyond expectations; however, the middle of cycle is non-limiting in terms of reactivity feedback. Agreement at the BOC and EOC is very good.
- Predicted worths for all banks throughout the cycle are in good agreement, as are the reactor trip characteristics used in safety analysis.
- PARCS calculated power feedback is negative.

Technical Topics

Section 4.4 – Thermal and Hydraulic Design

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Areas of Review

- Thermal and Hydraulic Design of the Core and RCS
  - Acceptable Analytical Methods used (largely based on data extrapolation from System 80+ and System 80)
  - Provides Acceptable Margins against conditions leading to fuel damage during normal operation and AOOs
  - Not Susceptible to Thermal-Hydraulic Instability
- Calculated Core Parameters to establish minimum DNBR are based on previously approved CE Methods
- Hydraulic Loads on Core and RCS during normal operation and DBA Conditions are extrapolated from System 80 flow test data
- Confirmed negligible differences in APR1400 core geometry and operating parameters to System 80+ and System 80 Designs

Technical Topics
Section 4.4 – Thermal and Hydraulic Design

Areas of Review

- Uncertainty analysis Methodologies namely Statistical Combination of Uncertainties to assure that at least 95-percent probability at 95-percent confidence level that hot fuel rod does not experience DNB during normal operation or AOOs
- CPCS Interfaces that support the DNBR and local power density (LPD) safety limits
- COLSS interfaces with the CPCS to support Chapter 7 reviews
- Computation of CPCS parameters needed for Core Reload
- Reactor protection systems design and operation, COLSS and CPCS, is essentially that of Palo Verde System 80 Design

Technical Topics
Section 4.4 – Thermal and Hydraulic Design

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Technical Report: APR1400-F-C-NR-12001-P "Thermal Design Methodology"

- CE Methodology based on approved TORC and CETOP codes with the KCE-1 CHF correlation
- CE Methodology approved statistical uncertainties methods used to assure a 95-percent Probability at 95-percent Confidence Level that the hot fuel rod does not experience DNB during normal operation or AOOs consistent with SRP including a penalty for rod bow

Technical Topics

Section 4.4 – Thermal and Hydraulic Design

Technical Reports: APR1400-F-C-NR-14003-P, "Functional Design Requirements for a Core Protection Calculator System for APR1400" and APR1400-F-C-NR-14002-P, "Functional Design Requirements for a Core Operating Limit Supervisory System for APR1400"

- CE Methodology based on approved methodology in use at Palo Verde, but also implemented at San Onofre, ANO-2, and Waterford since the early 1980's
- CPC protection software design assures with 95% probability and 95% confidence that DNBR and LPD limits are maintained
- Procedures for development of CPCS constants will be developed by the COL holder consistent with existing procedures for the operating System 80 plants

Technical Topics

Section 4.4 – Thermal and Hydraulic Design

Technical Report: APR1400-F-C-NR-14001-P "CPC Setpoint Analysis Methodology APR1400"

- CPC computes changes in linear power density and provides measurements of core conditions for peak power density and DNBR based on ex-core instrument measurements
- CPC uses these monitored parameters to calculate the LPD and DNBR margin to trip limits, Cycle-dependent uncertainties associated with the CPC trip point settings are combined such that the adjusted LPD and DNBR setpoints are always conservative

Technical Topics

Section 4.4 – Thermal and Hydraulic Design

Technical Report: APR1400-Z-J-NR-14004-P "Uncertainty Methodology and Application for Instrumentation"

- Describes methodology used to combine uncertainties to ensure plant protective functions activate at desired values under normal and accident conditions
- Essentially identical to approved CE topical CEN-356(V)-P-A (1988)

Technical Report: APR1400-Z-J-NR-14005-P "Setpoint Methodology for Plant Protection System"

- Describes methodology used to establish setpoints to be used for the ex-core PPS trip settings
- PPS functions contain the other RPS trips and the ESFAS actuation trips
- Setpoints established such that during DBEs analytical limit (AL) not exceeded. ALs are established such that safety limits (SLs) not reached, and SLs assure that unacceptable consequences do not occur during DBEs

Technical Topics

Section 4.4 – Thermal and Hydraulic Design

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Challenging Review Area: CPCS and COLSS Functions and Interfaces

- Basis for CPCS was not well documented in System 80+ DCD
- KHNP followed System 80+ DCD but did not link references to post-1980's functional changes and improvements to current system 80 design
- Staff concern: Basis documentation inadequate per 52.47(a) to assure safety
- Staff audit conducted January 2016
- The issue was ultimately resolved for the following reasons:
  - CE references located for CPCS and COLSS
  - Confirmed functionality are based on approved and implemented PVNGS changes

Technical Topics

Section 4.4 – Thermal and Hydraulic Design

Challenging Review Area: Single vs Double-sided Confidence Limit

- The statistical combination of uncertainties (SCU) methodology in the CPCS Setpoint technical report used to convert the DNBR safety limit of 1.124 to the analytical limit of 1.29 for 95/95 confidence level does not adhere to a recently updated RG 1.105 (requiring doubled sided, Rev 3 versus single sided confidence levels, Rev 2)
- Single sided Methodology based on CEN-356, approved by staff in 1989
- DCD Chapter 7 indicates methodology does not conform with RG 1.105 Rev. 3, so methodology used is not consistent with RG referenced
- Item remains open:
    - Based on CE operating plant experiences using CEN-356, staff will consider approval if the applicant takes exception to the setpoint methodology in RG 1.105 Rev. 3.

Technical Topics
Section 4.4 – Thermal and Hydraulic Design

Findings – Thermal-Hydraulic Design meets the following requirements:

- GDC 10: SAFDLs not exceeded during normal operation or AOOs
- GDC 12: suppression of reactor power oscillations can be reliably detected and suppressed

- Thermal design methodology depends heavily upon codes and methods previously approved by staff for domestic CE plants
- APR1400 thermal hydraulic design is comparable to System 80+ design with small differences due to the slight increase (i.e., < 2 %) in power of the APR1400
- Staff concludes that the design provides adequate assurance that the reactor will perform its related safety functions under all modes of operation pending completion of open items

Technical Topics

Section 4.5.1 – Control Rod Drive System Structural Materials

Technical Topics

Control Rod Drive (CRD) materials for reactor coolant system (RCS) pressure and non-pressure boundary components.

- KHNP provided:
  - Material selection, fabrication techniques, heat treatments and cleanliness control per NUREG-0800
  - Materials used for pressure boundary components are consistent with other designs and have satisfactory operating experience, such as stabilized stainless steel (Grades 347 and 348), martensitic stainless steel (ASME Code Case N-4-13 modified Type 403), nickel-based alloy (alloy 690, thermally treated) and austenitic stainless steel Type 304 and Type 316. Welding filler materials Alloy 52/52M, Alloy 152, and Alloy type 316L.
  - Materials used for non-pressure boundary components are consistent with other designs and have satisfactory operating experience, including austenitic stainless steels Type 316, Type 321, and Type 304. Martensitic stainless steels (Type 410 with conditions A and T, and Type 440C). Nickel-based alloys (Alloy X-750) and nickel-chromium-molybdenum-columbium alloy (Alloy 625).

Technical Topics
Section 4.5.1 – Control Rod Drive System Structural Materials

Technical Topics

- Use of austenitic stainless steel base materials is consistent with the recommendations of RG 1.44, and that only procedures that have been demonstrated to not sensitize CRD stainless steel components are used.
- Controls for abrasive work and cleaning on austenitic stainless steel surfaces are used to prevent cold work and contamination as specified in RG 1.28 and ASME NQA-1.
- Heat treatments consistent with NUREG-0800 and operating experience, including Alloy X-750 heat treated to 1149 degrees C (2100 degrees F), and Type 410 Condition T heated above 565°C (1050°F) for the ASME A276 Type 410T materials

Technical Topics
Section 4.5.1 – Control Rod Drive System Structural Materials

<u>**Open Items**</u>

- **RAI 8684, Questions 04.05.01-15** – The applicant needs to provide the material specifications and types for the Versa Vent™ component which is used for venting the CRD to minimize increased levels of oxygenated water in stagnant or dead end areas of the CRD components
  - Versa Vent™ is considered by the applicant as a non-pressure boundary component that replaces the CRD housing nut which is a pressure boundary component.
  - **Status:** The response was received and is in evaluation.

- **RAI 8684, Questions 04.05.01-16** – The applicant needs to provide data/operating experience that justifies venting during refueling outages keeps oxygen levels low (as stated in RG 1.44) so that non-L grade stainless steels can be used
  - **Status:** The response was received and is in evaluation.

Technical Topics

Section 4.5.2 – Reactor Internals and Core Support Materials

- Reactor Internals and Core Support Materials and Design in APR-1400 is consistent with industry practice. Applicant indicated a high level of adherence to appropriate controls, including adherence to relevant NRC Regulatory Guides. Staff RAIs focused on clarification and completeness. Staff found section acceptable pending incorporation of content from RAI responses.

Technical Topics

Section 4.5.2 – Reactor Internals and Core Support Materials

Staff review focused on the following topic areas:

- Materials specifications, selection, and heat treatments;
- Controls on welding;
- Nondestructive examination;
- Austenitic stainless steels;
- Other materials;
- Other degradation mechanisms (IASCC and void swelling in particular).

Technical Topics

Section 4.6 – Functional Design of Reactivity Control Systems

Areas of Review

- Control rod drive system (CRDS) functionality and arrangement
- Environmental and seismic qualifications of CRDS
- Design requirements for CRDS cooling system
- Possible single failures of CRDS
- Testing and verification of the CRDS
  - Initial test program
  - ITAAC and TS surveillance requirements for scram time
- Combined performance of reactivity control systems
  - Combination of systems used to mitigate specific AOOs and postulated accidents
  - Transient and accident analyses show reactivity control systems capable of controlling reactivity changes during design basis events (DBEs)
- Common mode failures of reactivity control systems
- TS requirements for reactivity control systems

Technical Topics
Section 4.6 – Functional Design of Reactivity Control Systems

Conclusions

- Meets GDC 4, 23, 25, 26, 27, 28, and 29 because:
  - Reasonable assurance that the CRDS will remain functional and provide safe shutdown capability under adverse environmental conditions and after postulated accidents
  - CRDS fails in a safe condition
  - No single malfunction in the CRDS will result in exceeding SAFDLs
  - CRDS and CVCS are independent, based on different design principles, and are capable of reliably controlling the rate of reactivity changes during normal operation
  - Transient and accident analyses show that the reactivity control systems can control reactivity changes during DBEs to maintain core cooling
  - Reactivity insertion limits prevent prompt power excursion
  - Extremely high probability of CRDS accomplishing safety function during AOOs
- Additional Considerations
  - CRDS cooling system meets design requirements
  - Initial test program adequate to verify reactivity control systems
  - Reactivity control systems TS ensure protection of public health and safety

ACRONYMS

AOO – anticipated operational occurrence
ASME – American Society of Mechanical Engineers
BOC – beginning of cycle
BOL – beginning of life
CE – Combustion Engineering
CEA – control element assembly
COL – combined license
COLSS – core operating limits supervisory system
CPC – core protection calculator
CPCS – core protection calculator system
CRD – control rod drive
CRDS – control rod drive system
CVCS – chemical and volume control system
DBE – design basis event
DCD – design control document
DNB – departure from nucleate boiling
DNBR – departure from nucleate boiling ratio
ENDF – Evaluated Nuclear Data File
ESFAS – emergency safeguards features actuation system
EOC – end of cycle
EOL – end of life
FTC – fuel temperature coefficient
GDC – general design criterion/criteria
ITAAC – inspections, tests, analyses, and acceptance criteria
LPD – local power density
MDC – moderator density coefficient
OPR1000 – Optimized Power Reactor 1000
PPS – plant protection system
RAI – request for additional information
RCS – reactor coolant system
RPS – reactor protection system
SAFDL – specified acceptable fuel design limit
SRP – Standard Review Plan
TCD – thermal conductivity degradation
TS – technical specifications