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UNITED STATES NUCLEAR REGULATORY COMMISSION'S
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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OPEN SESSION

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TUESDAY

APRIL 17, 2018

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ROCKVILLE, MARYLAND

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

COMMITTEE MEMBERS:

RONALD G. BALLINGER, Chairman

CHARLES H. BROWN, JR., Member

MICHAEL CORRADINI, Member

VESNA B. DIMITRIJEVIC, Member

JOSE A. MARCH-LEUBA, Member
DANA A. POWERS, Member
JOY L. REMPE, Member
PETER 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:
JEONG HYEON BAEK, KEPCO E&C
LAUREL BAUER, NRO
JOSEPH BRAVERMAN, Brookhaven National Lab
QUINTANA LUISS CANDELARIO, NRO
MARWAN CHARROUF, MPR
SUNGJU CHO, KHNP
WOOCHONG CHON, KHNP
JORGE CINTRON-RIVERA, NRR
JAMES GILMER, NRO
SYED HAIDER, NRO

NICHOLAS HANSING, NRO
MICHELLE HART, NRO
DAVID HEESZEL, NRO*
RAUL HERNANDEZ, NRO
ATA ISTAR, NRO*
JUNG BUM JANG, KEPCO E&C
JAEHOON JEONG, KNF
REBECCA KARAS, NRO
STORM KAUFFMAN, MPR
MOHSEN KHATIB-RAHBAR, ERI
GEESEOK KIM, KEPCO E&C
HAKSUNG KIM, KNF
TAE-HAN KIM, KEPCO E&C
YUNHO KIM, KHNP
YOUNGKI KIM, KEPCO E&C
WILLIAM A. KRIEL, KHNP
SUNGUK KWON, KHNP
YOUNGCHUL KWON, KHNP
SANG GYU LEE, KEPCO E&C
KAEYEOL LEW, KNF
CHANG-YANG-LI, NRO
Y.C. LI, NRO
DAEHEON LIM, KEPCO E&C
MARK LINTZ, NRO
SHANLAI LU, NRO

TIM LUPOLD, NRO
GREGORY MAKAR, NRO
MICHAEL D. MCCOPPIN, NRO
JILL MONAHAN, Westinghouse
ALISSA NEUHAUSEN, NRO
JINSUO NIE, NRO
JIYONG OH, KHNP
JUHYUN PARK, KHNP
SUNWOO PARK, NRO
ERIC REICHELT, NRO
ROBERT ROCHE-RIVERA, NRO
RICARDO RODRIGUEZ, NMSS
THOMAS SCARBROUGH, NRO
ROB SISK, Westinghouse
JOE STAUDENMEIER, RES
JAMES STECKEL, NRO
EDWARD STUTZCAGE, NRO
WALTER TAUCHE, ERI
TOMEKA TERRY, NRO
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MARIELIZ VERA, NRO
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JAY WALLACE, RES
WILLIAM WARD, NRO
YUKEN WONG, NRO
JUNG YONG ZAN, KEPCO E&C

*Present via telephone

T-A-B-L-E O-F C-O-N-T-E-N-T-S

P R O C E E D I N G S

8:30 a.m.

CHAIRMAN BALLINGER: The meeting will now come to order. This is a meeting of the APR1400 Subcommittee of the Advisory Committee on Reactor Safeguards.

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

ACRS members in attendance are -- we will have Vesna Dimitrijevic I think. Joy Rempe, Jose March-Leuba, John Stetkar, Matt Sunseri, Dana Powers, Dick Skillman and Mike Corradini. We may be joining -- let's see, Charlie Brown will probably be here shortly I guess.

Chris Brown is the designated federal official for this meeting.

The purpose of today's meeting is for the subcommittee to receive briefings from Korea Electric Power Corporation and Korea Hydro and Nuclear Power Company Limited regarding their design certification and the NRC staff regarding their safety evaluation report specific to chapter 2 section 2.5 geology, seismology and geotechnical engineering, chapter 3, design of structure systems, components, and equipment, chapter 7, instrumentation and controls,

chapter 15, transient and accident analysis, and topical report large break loss of coolant accident.

The ACRS was established by statute and is governed by the Federal Advisory Committee Act, FACA. That means that the committee can only speak through its published letter reports.

We hold meetings to gather information to support our deliberations. Interested parties who wish to provide comments can contact our offices requesting time after the meeting announcement is published in the Federal Register.

That said, we also set aside 10 minutes for comments from members of the public attending or listening to our meetings. Written comments are also welcome.

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

The rules for participation in today's meeting were announced in the Federal Register on Friday, April 13, Friday the 13th of 2018.

The meeting was announced as an open, closed to the public meeting. This means that the chairman can close the meeting as needed to protect

information proprietary to KHNP and its vendors. No requests for making a statement of the subcommittee has been received.

And we will be doing -- part of the meetings today will be closed so we'll be switching over.

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

When addressing the subcommittee participants should first identify themselves and speak with sufficient clarity and volume so that they can be readily heard.

There's a little black triangular looking thing in front of you that has a little light at the top. You need to make it light up green when you talk.

We have a bridge line established for interested members of the public to listen in and we have a second bridge line for KHNP and other people I think.

To minimize disturbance the public line will be kept in a listen-in only mode. The public

will have an opportunity to make a statement to provide comments at a designated time towards the end of the meeting.

I request now that meeting attendees and participants silence their cell phones and other electronic devices.

Also if you're on one of the bridge lines please mute your line unless you're making a comment or talking because it limits the amount of noise we have.

NRO staff and contractors are on a separate non-public bridge line with Member Riccardella. I apologize, Pete Riccardella is also going to be here.

Except for Member Riccardella I ask staff and contractors participating by phone to please keep your phone muted. I already said that.

Pete, are you on the line?

MEMBER RICCARDELLA: I'm here, but I don't think you can hear me.

CHAIRMAN BALLINGER: We can hear you now.

MEMBER RICCARDELLA: Okay, very good. Thank you.

CHAIRMAN BALLINGER: And before we get started Member March-Leuba needs to make a comment

related to conflict of interest.

MEMBER MARCH-LEUBA: Thank you, Mr. Chairman. In a previous life I used to work for NRO as a consultant on APR1400 matters especially on the corporate rotation calculator and cell points.

And if those items come along I will not participate in discussions.

CHAIRMAN BALLINGER: Okay. That being said now I invite Bill Ward is here hiding behind -- make a comment and start the meeting.

MR. WARD: Good morning. Thank you again for meeting with us. We are pleased to present section 2.5 three chapters in the large break LOCA topical report over the next two days.

I know we've been working at a very aggressive schedule and it's been a little bit frantic at times but we're happy to have the opportunity to present these and we appreciate the subcommittee's patience and willingness to work with us to get things done. So thank you.

CHAIRMAN BALLINGER: So I think the floor is yours.

MR. SISK: Thank you, Chairman. This is Rob Sisk, Westinghouse. And again on behalf of KHNP we do appreciate this opportunity to continue the

presentation of the APR1400 design certification application to the ACRS subcommittee.

Without further ado I will just turn the meeting over now to Mr. Kwon to get us started on the chapter 3 presentations.

MR. KWON: Good morning, ladies and gentlemen. I am Youngchul Kwon, senior structural engineer working for KEPCO Engineering and Construction Company. Now here we'll be presenting the work results for the open attempts related to APR1400 design control document tier 2 chapter 3.

This presentation material consists of overview of chapter 3 summary of open attempts, product.

This document tier 2 chapter 3 consists of 13 sections. Overview of chapter 3 is shown on this slide and next slide.

The submitted document related to chapter 3 are APR1400 design control document tier 2 chapter 3, design control document tier 1 and various technical reports shown on this slide and next slide.

The total 260 RAIs were captured in regards to chapter 3. Total number of open attempt is 23. And this will be addressed in this presentation.

This slide shows open attempts for chapter

3 with related RN numbers, topic names and their assessed numbers.

This slide also lists open attempts for chapter 3. And same.

Now we start to present the summaries of open attempts. There's no open attempt in sections 3.1 through 3.5. Staff tracked the RAI question 3.6.2-2 as an open attempt for the topic of pipe rupture hazard analysis.

In regards to these RAs, tested assessment of the dynamic effects of jet impingement and first wave. And considerations in determining which break locations in ASME class 1 and battery systems.

Staff also requested to provide technical report for pipe rupture hazard analysis, PHRA summary incorporating jet impingement, first wave and resident effects.

KHNP provided the PHRA summary report incorporating the post- break locations and the methodology for assessment of jet impingement and place waves in October 2017.

Also KHNP provided the text description regarding non-conservatisms over ANSI/ANS 58.2 1988 version in the revised response to this RAI in October 2017.

The staff tracked the RAI question 3.6.2-3 as an open item for topic of break exclusion area.

On this topic staff requested to provide acceptability of expanding the break exclusion area to the auxiliary building anchor wall beyond the isolation valve.

And requested to provide justification with the results of pipe rupture analyses.

KHNP provided confirmation that calculate the maximum stress results for the grade approach piping in break exclusion area were low compared to the SRP BTP 3-4 stress limit for postulated break locations in the revised response to this RAI in September 2017.

The staff tracked RAI question 3.6.2-6, 7 and 8 as an open item for the topic of blast wave and potential feedback implication and (inaudible) effect.

On this issue staff requested to provide clarification on the verification and validation, V&V, of the computational fluid dynamics CFD model for blast wave effects.

Methodology for evaluation of a blast wave and justification of the criteria used for considering the effect of a potential oscillatory jet loads.

As a resolution KHNP provided a technique

called report, APR1400 high energy line break with jet impingement containing CFD analysis, V&V results and the source of the literature resources in June 2017.

KHNP provided in technical report that the feedback implication and the resonance effects due to a high energy line breaks does not occur in APR1400 with thermodynamics and the lack of perpendicular flat surfaces.

During the review the staff requested additional information or clarification for evaluating the dynamic jet impingement and the blast wave effects to complete the staff's review.

KHNP provided the revised technical report including the clarification such as literature summary and CFD V&V in October 2017.

This slide shows the open item for leak before break analysis related to RAI question 3.6.3-9.

On this topic staff noted that the pipe crack evaluation program PICEP input did not agree with the values stated in the APR1400 design control document and requested to confirm the input values and calculations in the design control document are correct and to provide copies of the PICEP input files.

As a resolution KHNP revised the fluid

temperature in the PICEP input and calculation and provided the response with the revised input values and calculations.

These results were incorporated into associated design control document and technical report, leak before break evaluation of a surge line.

On the open RAI question 3.7.2.-1 for the topic of hard rock high frequency staff requested to provide -- create a number of modes to capture incoherent motion.

I'm sorry, missing the word incoherent motion -- to capture incoherent motion in print. And provide justification for ISRS reduction levels in excess of those provided in SRP section 3.7.2 subsection 2.4.

Also staff requested providing the revised results for ISRS comparisons and seismic loads between the certified seismic design response spectra, CSDRS, and the hard rock high frequency spectra and the justification for the acceptability of all HRHF exceedances.

In regards to this issue 16 modes was selected to capture, convert the incoherent motion. It was justified that this converged solution gave the ISRS reduction levels still exceeding the reduction

level set forth in SRP 3.7.2 as described in technical report evaluation of impacts of HRHF for response spectra on structure system components appendix C.

On the open RAI question 3.8.3-5 for the topic of analysis methods and the results for reactor containment building internal structure staff requested to explain the analysis method under seismic loading for floor slabs between the secondary shield wall and the containment shell, and to explain the connections between the floor slabs and containment shell.

Regarding this issue for the seismic load of a floor slab response spectrum analyses were performed using the FRS which envelop both containment shell and secondary shield wall. A detailed procedure for design and analysis of the floor slabs were described in design control document subsection 3.8.3.4.

And the design results were provided in DCD table 3.8A-43 and 44, and figures 3.8A-61.

Also KHNP provided connection details between the floor slabs and containment shell in design control document subsection 3.8A.1.4 and figure 3.8A-60.

This would allow the movement of both

horizontal directions in tangential and radial.

On open RAI question 3.8.5-1 for the topic of analysis and design for critical sections staff requested to include the missing information of other design sections and provided the lacking design results of concrete structures with a consistent format.

To resolve this issue KHNP provided the information of selection criteria for design sections in design control document section 3.8A and added the missing design sections, dome, liner, slab, steel beam and their design results in DCD subsection 3.8A.1.4.

KHNP provided the information of design results for concrete structures where the information was lacking. Accordingly, associated DCD subsections and tables for IRWST structure and the basement structure were updated.

On the open RAI question 3.8.5-4 for topic of waterproofing membranes staff requested to provide the description whether waterproofing membrane will use the APR1400 design and also requested to provide the design effects of the waterproofing membranes on shear resistance of a nuclear island the common basemat if they were used.

As a resolution KHNP confirmed the use of

waterproofing membranes in APR1400 design and provided the information for typical installation details as shown in design control document subsection 3.8.5.1 and figure 3.8-27.

KHNP also provided the sliding resistance effect associated with the waterproofing membrane beneath concrete basemat in the RAI response and the design control document subsection 3.8.5.1.

The design parameter of coefficient of friction will be verified by COL item number 3.8(11).

On the open RAI question 3.8.5-7 for the topic of a construction sequence staff requested to describe how the construction sequence and the differential settlement of foundations were considered in the load and the load combination and requested to provide the approach for four types of settlement, maximum vertical settlement, tilt settlement, differential settlement between structures and the angular distortion based on analysis results.

As a resolution the member forces and the moments from the analyses with and without construction sequence were considered in design load case of the design load combination. And the detailed description were added in design control document subsection 3.8.5.4.

For four types of settlement the approach for assessment and their results were provided in the DCD subsection 3.8.5.4. And technical report Stability Check for Nuclear Island Common Basemat section 5.

These four types of settlement will be verified based on the site-specific soil conditions by the COL item number 3.8(18) and the COL item number 3.8(19).

On the open RAI question 3.8.5-8 for topic of design and analysis procedures for nuclear island common basemat the staff requested to justify the use of equivalent static accelerations with a 100-40-40 combination method considering phasing motion of three superstructures, and requested to explain how to consider torsional load in the basemat analysis.

To consider phasing motion of three superstructures of reactor containment building, containment structures, reactor building integral structure and auxiliary building the results from both the linear SRSS combination on the member force level and the nonlinear method using 100-40-40 combination on the input level were enveloped.

Additional detail descriptions were added in DCD subsection 3.8A.1.4 and technical report

Stability Check for Nuclear Island Common Basemat section 3.2.5.

For the member forces and the moments from torsional analysis of superstructures they were added to the seismic load cases by absolute summation. Additional detailed descriptions were added in the above mentioned design control document and technical report.

On open RAI question 3.8.5-12 for the topic of applied loads for nuclear island common basemat staff requested under the KHNP response to RAI question 3.8.5-8 to provide the maximum bearing pressure resulting from both the linear SRSS combination and the nonlinear 100-40-40 combination to resolve phasing of the responses from each of the three superstructures when analyzing nuclear island common basemat.

For bearing pressures for nuclear island the common basemat they were recalculated from both the linear SRSS combination and the nonlinear 100-40-40 combination. Enveloped bearing pressures were provided in design control document subsection 3.8A.1.4.

MEMBER STETKAR: Mr. Kwon?

MR. KWON: Yes.

MEMBER STETKAR: And if we delve into proprietary information we can wait.

I'm not a structural engineer but I noticed that when you did those recalculations the maximum bearing pressures changed substantially.

For example, the static bearing pressure in DCD revision 1 was about 642 kilopascals and it was revised to 937 kilopascals.

The bearing pressure with the applied earthquake loads in revision 1 was 1,416 kilopascals and it was revised to 2,586.

To me not being a structural engineer those sound like rather substantial changes. Could you tell us what was changed in the analysis to result in those large changes?

You said that you recalculated the loads.

MR. KWON: The division of --

MEMBER STETKAR: I understand. I'm asking what changed in your analysis to result in those rather substantial revisions.

As I said if there's something proprietary that we need to discuss in a closed session I'm willing to wait. But if it's something simple.

I did not read both versions of the RAIs and the technical reports to try to delve into it nor

do I necessarily have all of the detailed structural engineering capabilities myself.

I found those changes to be rather substantial.

MR. SISK: This is Rob Sisk. They were just talking a little bit about what changed. We'd have to go back to the documents in detail to give you.

MEMBER STETKAR: Fine. Thanks. I'm happy with that. Thanks.

MR. KWON: I'll check the changes. On the open RAI question 3.8.5-13 for topic of loads and load combinations for nuclear island common basemat staff requested to confirm that the crane loads are considered in the design of the entire containment and other seismic category 1 structures, and confirm that the crane/trolley load is considered in different positions to maximize the loads on the structures.

The crane loads were considered in the design of the containment structure and other seismic category 1 structures.

The positions to give maximum polar crane loads in the design of containment structures were described in design control document subsection 3.8.1.3.

In case of auxiliary building the enveloped crane loads from various fuel handling area crane locations were considered and described in design control document subsection 3.8.4.3.

The parking positions of the actual polar crane and trolley will be verified by COL item number 3.8(21).

On open RAI question 3.8.5-16 for topic of subgrade modulus of soil profiles for APR1400 staff requested to update stability evaluations in all applicable parts of design control document sections 3.8.5 and 3.8A and technical report to reflect the current models, analysis approach and the results.

Also staff requested to provide the descriptions for bearing pressure evaluations in the emergency diesel generator building and the diesel fuel oil tank structures.

Regarding this item the updated basemat on the stability evaluations applying the distributed soil spring elements for static conditions and the solid soil elements for dynamic conditions were added in design control document subsection 3.8A.1.4.

And the technical report titled Stability check for Nuclear Island Common Basemat section 4.2.3 and table A-7.

And the description for bearing pressure evaluation were added in design control document subsection 3.8A.1.4 for nuclear island common basemat and subsection 3.8A.3.4 for emergency diesel generator building and diesel fuel oil tank basemats respectively.

On open RAI question 3.8.5-17 for the topic of a differential settlement of nuclear island common basemat staff requested to explain the determination method of differential settlement.

The differential settlements were determined based on the construction phase and the post-construction phase. And a detailed procedure was added in the design control document subsection 3.8.5 and technical report Stability Check for Nuclear Island Common Basemat section 5.

Next open item will be presented by Mr. Kim.

MR. GEESEOK KIM: Good morning. My name is Geeseok Kim. I'm working for KEPCO E&C as a mechanical engineer.

This slide showed open item for the pump-induced load on reactor vessel internal components.

The staff requested to provide the test on application of the measured pump pressure pulsations

and related hydraulic model. Also to request to add a description of the DPVIB computer program to DCD.

KHNP provided information on methodologies for application of the measured data. The methodologies are prequal to obtain functions for all configurations from measured data.

Particularly KHNP submitted a PNV document for DPVIB computer program.

This slide shows the open item for control element guide tube buckling integrity and operating experience.

The staff requested regarding the buckling integrity of the control element guide tube under Level D conditions and requested that operating experience in Korea.

Theoretically if the guide tube is deflected it cannot buckle. Therefore, buckling does not occur in the guide tube. But the buckling integrity of the CEA guide tube was evaluated and the detail calculation was submitted.

KHNP reviewed operating history in Korea domestic plants and confirmed that there are no historic damage in the fuel assembly.

This slide shows the buckling analysis and the position (inaudible) in the tube.

The analysis of the CEA guide tube is evaluated in accordance with ASME section 3 and subsection NGA 3211 and NGA 3133.

The evaluation is a proponent for the condition. Actually the movement of tube is shown back at the figure. The tubes are raised up, both sides, you just put plate.

However, pure (inaudible) plate is not fixed and the tube could be translated at the direction.

Using the ASME 1334 critical buckling are determinator. This critical load missed the limit of the buckling.

Also we can load, the load of the tube is calculated using the stress of the tube which is calculated from the stress report summary.

Therefore, finally we check the CEA guide tube. It is not susceptible to buckling.

We go to the guide tube tolerance. One of the functions of the CEA is that mechanical clearance of the CEA and the fuel internals are such that the shared positioning and the reactor trips are obtained.

The accumulation of tolerance. These are shown in DC section 4.2.1.

Also the second sentence show the

alignment check provides reasonable assurance that the frictional force that could result from adverse tolerance is below the force that could significantly increase scram time.

This sentence is shown in DCD section 4.2.4.

MEMBER REMPE: I have a question. It's a curiosity question. My understanding when I was reviewing the material is that this RAI was motivated because swelling was observed in the control tips at Palo Verde.

And I was -- well, first of all I was curious on what materials were in the control tips at Palo Verde and then I was expecting -- I know that the RAI response said we searched through and we've never had this happen with any of our Korean plants.

But I was expecting to ensure that this did not happen if an APR1400 was built in the U.S. you'd say don't use that material.

And so could you give me a little background so I could understand just for my own curiosity on what material was used at Palo Verde that did have that swelling?

MR. SISK: This is Rob Sisk. I don't think we can comment on what was used or what's not

used in Palo Verde so I'm not sure we could really comment on that.

MEMBER REMPE: Well, if there was a problem in the past, the swelling of a particular material, I think it would behoove us to all remember what it was and it doesn't get used again.

And so I think it's a reasonable question. Do you know, Ron, what it was just from your own background?

CHAIRMAN BALLINGER: No.

MR. SISK: Again, Rob Sisk. We could check into that. Our fuel people are actually here and we can discuss that with them later.

MEMBER REMPE: Okay. Perhaps the staff knows too, I don't know. But I'm curious on that just for my own knowledge. Thank you.

CHAIRMAN BALLINGER: Sorry for the interruption but we're joined by Member Dimitrijevic and Member Brown.

MEMBER SUNSERI: Maybe just to add to Joy's comment. There was an industry problem control rods made with hafnium. And that material swelled and it -- especially at the tip of the rods where they were close to the core.

And the fix was to eliminate the hafnium

and go to silver-indium-cadmium and taper the tip a little bit so it would allow for some growth. But I don't know if that applies to this unit or not.

MR. SISK: We're aware of that as well. That's fairly old OE. I don't want to say anything until we really kind of check back with the right people.

MEMBER REMPE: If you would. For my own knowledge to know what it was. Thanks.

MR. GEESEOK KIM: The third sentence also to confirm the instability of the CEA, the CEA positioning test is performed in every reporting period. According to the requirements and technical specifications for section 3.1.4.

This slide shows open item for the in-core instrumentation support system.

The staff requested to provide information about static O-ring seal for the inquiry into instrumentation system which inform the rate of coolant for the boundary.

The staff commented that the summary of this information including clarification and the (inaudible) should be included in the DCD.

KHNP provided explanation for the classification, design requirements and the design

conditions of the static O-ring seal in the RAI response.

Also reflected this item in revision 1 of the DCD section 3.9.5.

KHNP also provided test result information including functional and physical properties to ensure that reactor pressure boundary is maintained throughout service life.

The static O-ring seal are replaced in every reporting period.

This slide shows open of the loading condition level D service loading.

The analysis --- also, the staff requested to describe and justify why difference between DCD tier 2 and table 3.9-1 and the loads applied to the reactor pressure vessel internals for all service levels.

The major difference between DCD tier 2 and table 3.9-1 is that IRWST in containment 30, the dirty water discharge loads are described in the loads and the load combinations in the DCD.

But table 3.9-1 shows the only title of transient events.

So KHNP provided explanation for the events that resulted in IRWST injection and discharge

loads in DCD.

This slide shows open item for the bases for reactor internals classification.

The next step also requested to provide the information of the non-classified seismic category 1. And to complete classification list in DCD.

KHNP provide a response. All of the reactor internals are classified as seismic category 1. The classification of the (inaudible) including internal provided in tier 2 table 3.2.-1.

KHNP other response to NRC issued to RAI 7927 and HR20 pertain to issues with the table 3.2-1 and the system quality group clarification related to reg guide 1.26 to not include the changes pertaining to this RAI.

Currently the complete clarification that is including internal structure is incorporated in the DCD revision 2 table 3.2-1.

This is an open item for the pump and the valve inservice testing program table.

The staff requested to evaluate and modify all pumps and valve in table to be consistent with NRC regulations in 10 CFR 50.55a.

KHNP added additional information of inservice test requirements for pumps and valves in

the RAI including our reports the revision to DCD table 3.9-13.

This slide is regarding environmental qualification of the mechanical and electrical equipment.

Staff requested to provide justification why in IEEE standard 323-2003 is acceptable for qualification of class 1E electrical equipment in the harsh environment.

In this respect the staff -- actually the IEEE standard 323 1974 version remain. It currently stand out.

And it has been endorsed by reg guide 1.89 for environmental qualification.

Nevertheless KHNP is about to apply IEEE standard 2003 version as the applicable standard for environmental qualification.

KHNP provided justification for the use of the IEEE standard 2003 stating that it conforms with 10 CFR 50.49 and there are no technical differences between the 2003 and 1974 versions.

The IEEE standard 2003 version reflects current practices for environmental qualifications. KHNP provided a basic table comparing the guidance contained in IEEE standard 1974 and 2003 versions.

This slide is regarding the environmental qualification of mechanical and electrical equipment.

The staff requested KHNP to clarify that qualification documentation will provide auditable records that show the equipment can perform its safety function during and following a design basis event as applicable.

Second, the staff requested KHNP to discuss the other applicable design event conditions and how the design is demonstrated to conform to 10 CFR 50.49.

The staff requested KHNP to clarify the discrepancy between IEEE 323-1974 and 2003 versions on the synergistic effects and explain how synergistic effects are considered for type test qualification in IEEE 2003.

KHNP provide the response as follows. First KHNP justified the documentation we provide audit report that show the equipment can perform its safety function during and after the process begins.

By clarifying this application based on IEEE standard 2003.

Second KHNP responded that the other accidents do not result in limiting conditions for safe shutdown consideration such as loss of coolant

accident, main steam line break and high energy live break, and the design is demonstrated to conform to 10 CFR 50.49.

Third, KHNP responded there are more discrepancy for synergistic effect between the two standards because the synergistic effects have already been considered in IEEE 2003 version.

And justified how synergistic effects are considered for test qualification by explaining the sequence of the age conditioning specified in IEEE 2003 version.

CHAIRMAN BALLINGER: Excuse me, but somebody out there does not have their phone on mute and we're getting shall we say unusual noises. So please put your phone on mute.

MR. SCHULTZ: Mr. Kim, could we go back to the previous slide. Yes, you've done that. Thirty-one, slide 31. No, I'm sorry. Thirty-three.

I don't understand the third bullet here where you described the IEEE standard and you've described what is expected in that standard. I'm just not sure in response to the RAI how you -- what your response was.

In other words have you then taken this into account in your evaluation and therefore the

issue is resolved or could you elaborate on that third bullet and what you've done with the information?

MR. GEESEOK KIM: I am not actually amazing in this area.

MR. SCHULTZ: Perhaps we can discuss it with the staff. I didn't see in the staff's presentation that they were going to go into this level of detail.

MR. SISK: This is Rob Sisk. Just for clarification to make sure I understand the question being asked your question is related to how age conditioning was applied.

MR. SCHULTZ: I think so. Again I'm trying to understand the sequence of information that's presented on the slide.

In the third bullet it's just a declaration that this is what the expectation is. And I was looking for if you will the step that demonstrates that it's been included or that it has been considered and therefore the issue is addressed.

Or if it's just being provided for information. It seems like another statement needs to be provided to say and it was considered rather than these are the things that need to be considered. That's all I was looking for.

If you need to get back that's fine.

MR. GEESEOK KIM: Yes, I will check. This slide is regarding the structural integrity of the piping and the pipe supports during the operation of the safety injection tank and the fluidic device.

Staff requested three items to clarify how cavitation effect and vibration originating from the operation of the safety injection tank and its fluidic device, how we take into account in the structural design evaluation of the safety injection tank, its discharge piping and pipe supports.

Secondly, to clarify whether the operation of the safety injection tank with its fluidic device can result in other phenomena such as water hammer.

And third, to clarify the structural evaluation model of the safety injection tank is coupled with the fluidic device.

KHNP --

CHAIRMAN BALLINGER: Can you go back to the previous slide? Again I guess I'm now more sensitized to the kinds of questions that Steve asked.

But the last bullet, the structural evaluation model of the SIT is coupled with the FD.

What does that mean? Does that mean the issue is taken care of because they're coupled?

MR. GEESEOK KIM: Yes.

CHAIRMAN BALLINGER: That's what you're saying.

MR. GEESEOK KIM: Yes.

CHAIRMAN BALLINGER: Okay, thank you.

MR. GEESEOK KIM: KHNP provides the following.

First, the cavitation noises are non-periodic and their frequencies are much higher than the modal frequencies of the safety injection tank and the safety injection line.

Therefore there is no possibility of resonance due to the cavitation implosions.

Second, the APR1400 safety injection system is designed to prevent the water hammer by eliminating possible void trap in the system based on the NUREG/CR-6519.

However, based on NRC feedback, the pressure and the dynamic forces on the piping system due to safety injection tank blowdown effect are calculated. The results show that all pipe stress meet code allowable and the support can be acceptable in existing support design.

Third, the structural evaluation model of the safety injection tank for seismic analysis is

coupled with the fluidic device because it is installed inside of the tank with the welding.

Currently chapter 3 is completed. KHNP continues to monitor chapter 3 to assure any conforming changes are addressed.

Twenty-three open items that were identified in phase 2 and 3 have been resolved with adequate and sufficient discussion with the staff.

The changes in chapter 3 as reviewed and marked up in response to the RAIs were incorporated into the revision 2 of the design control document.

MR. SISK: So this concludes the presentation on chapter 3. We can move into 2.5 at this point or if there are questions.

MR. SCHULTZ: I have one question. Mr. Kwon, this will lead you back to the beginning of your presentation but since we're going to talk about large break LOCA this afternoon I wanted to get just a little more information on your response to 3.6.2 dynamic effects in piping and 3.6.3 on leak before break.

In the staff's evaluation of the RAI and in discussions that they had with you one thing that they noted that wasn't in your presentation today was the benefit of the inspection program capability that

you have for piping and weld inspection.

And I wondered if you could describe a little more about what your expectations are for in-service inspection of piping and welds especially the large piping and welds.

This is with respect to especially the dynamic effects on piping. One of the things the staff mentioned in their discussions associated with the RAI and your response had to do with the expectation that you have capability for 100 percent inspection of piping and welds and I just wanted to get confirmation of that, that you had set that as expectation in the operation of the facility.

MR. GEESEOK KIM: You mean that the piping (inaudible)?

MR. SCHULTZ: Yes, well the inspection program can affect what one's findings would be with respect to the dynamic effects of piping and leak before break.

So I just wanted to get your confirmation of what your expectations would be for an appropriate inspection program of piping and welds.

MR. GEESEOK KIM: Inspection is performed, as you know. It is a component on the average of 10 years. Every bit for major piping but is that

benefits and also to control that. Almost all classes including class 1 is inspected. Through the inspection, most of the --- vibration -- piping system. That's the inspection.

MR. SCHULTZ: Those are the preoperational testing. And then you have your 10-year inspection requirement.

MR. GEESEOK KIM: Yes.

MR. SCHULTZ: You do have excess ability for 100 percent inspection in 10-year intervals. Thank you.

MEMBER REMPE: So is this something that's passed on to the person that's building the plant and so is there like an ITAC that identifies all the requirements for the in-service inspection as part of the design certification?

Where's the boundary on how you convey what you want the person building the plant to do an in-service inspection?

MR. YUNHO KIM: I'm Yunho Kim from KHNP. Actually we don't have the right person to answer that question.

MEMBER REMPE: I can't hear you even though you're talking into a microphone. Say it again real loud.

MR. SISK: It's on. Rob Sisk. We don't have the right people here to discuss the details of the inspection program.

In general I will say the inspection program tends to be a utility program.

MEMBER REMPE: Right.

MR. SISK: So it's a combination of both the designer and the utility.

CHAIRMAN BALLINGER: It's an ASME code issue. So section 11 in particular defines the inspection intervals for everybody.

MEMBER REMPE: Okay.

MR. SISK: Any other questions on chapter 3? We'll go on to 2.5.

This is Rob Sisk again. While they're getting ready for 2.5 we did have a follow-up discussion or we would like to have a little bit of a follow-up on the question that came up in chapter 3 on what changed in the analysis. I think that was Mr. Stetkar's question. I invite Youngki Kim to provide some additional details for us.

MR. YOUNGKI KIM: Youngki Kim from KHNP. I explain about the reason why the bearing stress was somewhat different compared to the report of the original response. Report of the original response,

100-40-40 method, used just combination of A, incorporating NRC question for combination number was increasing to 96. So combination increases potentially. So we think that's the reason why the bearing stress was increasing so much.

MEMBER STETKAR: I didn't quite understand it. But why did you -- what caused you to change the load that much. If I understand your response.

MR. YOUNGKI KIM: You apply 100-40-40 in combination of stress and the pressure. So you combine the load. Combinations to part of the critical stress. So I was told that, before the other response, you had only eight combination cases when you apply another vessel following, the combination number is increased from 8 to 96. We think that's the reason for the increase of the bearing stress.

MEMBER STETKAR: Okay. Maybe I'll follow up with the staff. Thank you. That helps a little bit. Thank you.

MR. SISK: So we're ready then to proceed onto section 2.5.

MR. KWON: Youngchul Kwon, again. I will present results for an issue item related to APR1400 design control document tier 2 chapter 2, section 2.5.

This material consists of an overview of

section 2.5 summary of issue item, current status and attachments.

Design control document tier 2 chapter 2 section 2.5 is described about geologic, seismologic and geotechnical site parameters for APR1400 plant design.

A list of submitted documents related to section 2.5 is shown on this slide. Total of 21 RAIs for chapter 2 questions and response. No open item.

In section 2.5 there is only one issue item for topic of static stability of foundation.

In regards to this topic staff requested to provide bearing pressures and the settlements of design control document section 2.5.4 per the related open RAIs question 3.8.5-7 and 16 of design control document section 3.8.5 when they were resolved.

Based on the response of RAI for DCD section 3.8.5 KHNP provided the revised static/dynamic bearing pressures for nuclear island common basemat, emergency diesel generator building and diesel fuel oil tank in design control document section 3.8A.1.4 and 3.8A.3.4.

The bearing pressures will be verified by evaluating with allowable bearing CPCS based on the site-specific properties by COL item number 2.5(13).

Also KHNP provided the revised maximum differential settlements inside buildings and between buildings in the design control document table 3.8A-17 and 39 and design control document section 3.8A.3.4.

For site suitability determination the predicted settlements will be verified whether they exceed the maximum settlement specified in design control document table 2.0-1 site parameters by COL item number 2.5(14).

Currently DCD chapter 2 section 2.5 is completed. KHNP continues monitoring section 2.5 to assure any conforming changes are addressed.

Changes in section 2.5 as reviewed and marked up in response to RAIs were incorporated into revision 2 version of the design control document.

MR. SISK: That concludes our presentation on section 2.5. Very short and sweet.

CHAIRMAN BALLINGER: Thank you. At the risk of jinxing we're very far ahead and I think the staff is ready to go. So I think we should just keep going.

MR. ROCHE-RIVERA: So the slides just presenting the technical staff involved in the review of chapter 3. Specifically in this case we'll be presenting the staff review for section 3.7 and 3.8.

So my name is Robert Roche-Rivera and I'm a structural engineer in NRO. And I am the reviewer for DCD section 3.7.2 which documents the applicant's seismic system evaluation.

Specifically this section of DCD addresses the seismic analysis methods and models which are used to establish the seismic demands for the design of the seismic category 1 structures of the APR1400 standard plan.

In the next few slides I'll be presenting the staff's evaluation of the applicant's seismic system evaluation to the hard rock high frequency spectra.

So the seismic analysis and design of the APR1400 standard plan is based on the certified seismic design response spectra or CSDRS.

Additionally the APR1400 is evaluated for the effects of the hard rock high frequency, called HRHF.

The HRHF spectra is a spectra that's represented before a number of rock sites in the central and eastern United States.

It has higher spectral amplitudes past about 10 hertz that's when it's compared to the CSDRS and it peaks at about 25 hertz.

In performing its evaluation the applicant took into consideration the incoherency of ground motion which reduces the in-structure responses such as in-structure response spectra in the high frequency range.

The staff review of the applicant's evaluation found that it lacked technical justification with respect to the number of spatial coherency modes used in the evaluation. The ISRS or in structure response spectra spectral amplitude reductions that were greater than the limits set forth in SRP section 3.7.2.II.4 and the adequacy of the structures for withstanding the seismic demands induced by the HRHF spectra.

To address these issues the applicant reevaluated the HRHF effects on the APR1400 standard plan with consideration of more than twice the original number of spatial coherency modes.

The applicant demonstrated the adequacy of the number of modes considered in their reevaluation based on comparisons of ISRS for alternative spatial coherency mode sets.

That is the revised set used in their reevaluation and a set with reduced number of spatial coherency modes.

And comparisons of ISRS showed really close agreements for these alternative coherency mode sets.

Based on such demonstration that an adequate number of coherency modes have been considered in the hard rock high frequency evaluation the staff found the associated reductions with the re-analysis to be acceptable.

And lastly the staff found the APR1400 structures to be qualified for the seismic demands imbued by the hard rock high frequency spectra based on higher CSDRS spectral accelerations relative to the hard rock high frequency spectra throughout the dominant frequency range for structure, higher equivalent accelerations that are used in the structural design obtained from the CSDRS which is the design basis for the APR1400 standard plan, and the applicant's demonstration that the reinforcement based on the CSDRS, again the design basis for the standard plan, it's adequate for the hard rock high frequency demand.

That concludes my presentation and with that I will turn it over to Mr. Vaughn Thomas.

MR. THOMAS: Thanks, Robert. Good morning. My name is Vaughn Thomas and I'm the

structural in the Office of New Reactors.

For the next few minutes I'm going to present the staff's safety findings related to the analysis and design of the operating concrete floor slabs which are part of the containment structures for the APR1400 application.

The operating floor slabs are located between the secondary shield walls and the containment walls.

Those slabs are supported by steel beams and beam seats attached to both sides of the walls. The staff focused its structure review on the design approach and methodology to ensure that the applicant's design for the CIS is reasonable and acceptable in terms of the scope, the level of details and the technical accuracy of the information provided by the applicant.

The staff found the review in accordance with agency regulatory requirements and staff guidance.

In reviewing the analysis methods and results associated with the design of the concrete internal structures the staff noticed that the DCD did not describe in detail the design analysis procedures and the decoupling criteria for the operating floor

slabs.

So as a result the staff requested additional information to ensure that the structural integrity of the concrete floor slabs are maintained in accordance with the agency's regulatory requirements.

To resolve this issue the applicant performed additional calculations and sensitivity analysis of the concrete floor slabs following the decoupling criteria in SRP section 3.7.2.

In other words applicant performed separate LOCA analysis of the design of the operating floor slabs.

And applicant's results demonstrated that the gap which is needed to prevent interaction between structures of 2 and 1/16 of an inch is adequate to allow the relative displacement between the containment internal floors and the containment wall.

And that 2 and 1/16 of an inch is greater than the maximum displacement of 2.04 inches.

MR. SCHULTZ: What was the nature of the sensitivity studies that they performed?

MR. THOMAS: The reason for that was that when we review the application we noticed that between the containment wall and the secondary shield wall

there were some slabs there.

We did not see any description or analysis in the documentation that says how they intend to address this particular walls or slabs.

So we ask them the question is it -- is a wall as part of the final analysis or -- and they said that well, they decided to design the walls separately. And they thought our acceptance criteria in 3.7.2 in terms of mass ratio and frequency ratio because it's relatively small.

And that's all the analysis they did to perform that.

MR. SCHULTZ: The sensitivity analyses were part of that? In order to incorporate the information on these additional slabs. Or is it all one piece?

MR. ROCHE-RIVERA: No, it is separate. Separate.

MR. SCHULTZ: Okay.

MR. ROCHE-RIVERA: And that's what I'm going to the next bullet that's about --

MR. SCHULTZ: Okay.

MR. ROCHE-RIVERA: It says the applicant re-analyzed the entire CIS including the primary shield wall, steam generator wall and the refueling

pool wall and the pressurizer wall using the reaction forces obtained from the LOCA analysis, that's for the concrete floor slabs.

The end result is increasing member forces and rebar arrangements for all those structures that are part of the containment internal structures.

So as a result the applicant also provided additional markups, figures, tables and text that describes in detail the design analysis approach of the operating floor slabs with the new design results of the CIS.

So in conclusion the staff considered the information provided by the applicant, demonstrated that the structural integrity of the CIS is adequately maintained following the agency's regulatory requirements.

So currently this issue is being tracked as a complementary item pending the next revision of the APR1400 DCD.

This concludes my presentation and if you do not have any questions I will turn it over to Ata Istar who is joining us over the phone to continue.

CHAIRMAN BALLINGER: I do have a question.

MR. THOMAS: Go ahead.

CHAIRMAN BALLINGER: Can we go back.

Whenever I read about concrete construction and I see a number like 1/16 of an inch I just know that's not correct.

So what is too small in other words? Two and 1/16 inch gap, what is too small?

MR. THOMAS: Okay, this is how it comes about. So when they did analysis of the CIS itself a maximum displacement for the CIS itself was 2.04 inches.

CHAIRMAN BALLINGER: Okay, so now I'm asking the question 0.04.

MR. THOMAS: Right. Analysis. Results. Analysis, results, that's what it shows.

CHAIRMAN BALLINGER: This is a TI87 calculation. Okay. And so both of those numbers are basically artificial.

MR. THOMAS: It's what the analysis results shows in the final analysis, yes.

CHAIRMAN BALLINGER: Okay, I'm just curious as to whether or not you actually have overlap in theory. Not in theory, excuse me, in practice. Whether you can have an overlap.

MR. THOMAS: So this is one reason we look at it is again they perform that separately. And when you run the analysis for the CIS overall it shows a

maximum displacement.

CHAIRMAN BALLINGER: I understand all that, what you do, but I'm just concerned about what you do with an answer. And then you ask about what's the uncertainty on that.

You've concluded that you're okay based on the calculation, but if you applied -- if you asked yourself what's the uncertainty on those two numbers do you in fact get overlap.

MR. THOMAS: Keep in mind that still following the regulatory guidance for example still within codes 349.

CHAIRMAN BALLINGER: Okay, thank you.

MR. THOMAS: Okay. Ata is on the phone so Ata, are you there?

MR. ISTAR: I'm here. Can you guys hear me?

MR. THOMAS: Yes.

MR. ISTAR: Good morning. I'm structural reviewer for section 3.8.5 which is foundations of seismic category 1 structures.

And I will go through some of the selected questions that the applicant responded and resolved and the markups provided in the DCD and the topical reports.

Starting with the critical section. The applicant did not capture all of the critical sections following their criteria.

The applicant added the missing critical sections in the DCD including structural steel and reinforced concrete sections.

And the staff determined that the design of these sections were consistent with the SRP 3.8 and the applicable codes and standards.

Next page please which is related to the waterproofing.

During the review it was not clear to the staff whether the waterproofing membrane was used for the foundations.

The applicant responded that the waterproofing membranes will be used for exterior below grade horizontal and vertical surfaces of the structures in APR1400 design with minimum coefficient of friction of 0.55.

The applicant has also assumed a coefficient of friction of 0.55 in the stability evaluation which is to be verified by the COL item 3.8(11).

Going to the next page related to the construction sequence and differential settlements.

The applicant did not provide any description for the construction sequence and differential settlement of nuclear island foundation.

And the applicant incorporated detailed explanations of the construction sequence and clearly determined the criteria for four settlement types which are maximum vertical settlement, tilt settlement, differential settlement between buildings and angular distortion for the nuclear island foundation.

CHAIRMAN BALLINGER: Excuse me, can you go back one slide?

MR. ISTAR: Yes, sir.

CHAIRMAN BALLINGER: The third bullet. How is the applicant going to verify that the friction coefficient is 0.55?

MR. ISTAR: I don't know how to respond to that one, but that is the value that was provided by the KHNP. And we have to abide with that number.

And how are they going to do it, that's during the construction sequence or construction of the plant. I believe that somehow they have to prove that the minimum value of coefficient of friction is 0.55.

MR. BRAVERMAN: Excuse me, my name is

Joseph Braverman from Brookhaven National Laboratory and we provide technical assistance to the staff in this review.

That would be a COL information item and so the COL applicant would have to provide documentation. Typically that's done by test data.

CHAIRMAN BALLINGER: Okay. So it's an actual test. Because this is a hard number that affects everything.

MR. ISTAR: I think I'm going to nuclear island basemat analysis.

It was not clear to the staff how the seismic and other loads are determined and applied to the various structures in the APR1400 basemat design.

And the applicant responded for the seismic loads the equivalent static analysis methodology is used and the envelop of the results of the linear case (SRSS combined method) and the nonlinear case (100-40-40 combined method) for three soil profiles, soft, medium and stiff, are used in the analysis of nuclear island basemat.

The staff finds this response acceptable because the applicant used the equivalent static analysis methodology and combined methods for the design of nuclear island basemat consistent with the

guidance provided in regulatory guide 1.92 and SRP 3.8.5.

Going into the last open item, maximum bearing pressure at nuclear island basemat.

The applicant was requested to determine the maximum soil bearing pressure for the nuclear island basemat for linear and nonlinear dynamic load combinations based on the updated analysis.

The applicant calculated the maximum soil bearing pressure for all load combinations (static, nonlinear, linear and SASSI analysis cases) and determined that they are lower than the allowable static and dynamic bearing pressure CPCS.

This will conclude my presentation and if you have any questions.

MEMBER STETKAR: I don't know if you were listening in when we had the presentation from the applicant, but I noted that when they re-performed the analyses the stated static maximum -- the maximum bearing pressure under static loading conditions increased apparently from about 642 kilopascals to 937 kilopascals.

And the maximum bearing pressure under seismic loads in combination with the static loads increased from about 1,416 kilopascals to about 2,586

kilopascals.

I also preface this by saying I am not a structural engineer but those changes struck me as being rather large.

Could you explain to not a practicing structural engineer what changed in their analyses to result in those rather large differences?

MR. ISTAR: I think I remember that discussion earlier, but I think the reason for that is the bearing area under the foundations is the most critical area.

And there was lengthy discussions about the bearing area, change in the bearing area. And if you were to pick up the right bearing area you're going to have a different bearing pressure.

MEMBER STETKAR: So they in their initial analyses did not have the correct bearing area, is that what I'm hearing you say?

MR. ISTAR: I don't want to conclude like that, but it was a lengthy discussions during our meetings with the staff, KHNP staff, engineering staff.

I think it was concluded that the correct bearing area had to be -- were selected for the loading conditions that was applied. And appropriate

bearing pressure was calculated based on that.

MR. BRAVERMAN: Can I add something to that. This is Joe Braverman again from Brookhaven Lab.

The staff's review of the original design they raised some questions which I think led to the increase in results.

The first one is the type of seismic analysis performed. This is for the basemat design. And there are three superstructures that sit on top, the containment, the internal structures and the auxiliary building.

So apparently the applicant used different types of analysis. For the containment and the internal structures they used response spectrum type analysis and for the auxiliary building they used equivalent static.

So in the response spectrum analysis you lose the sign or the phasing whereas equivalent static you apply a G value one direction so you know the sign of the defamation of it, et cetera.

The applicant used the reaction forces at the bottom of these superstructures and applied these forces onto the basemat. So the staff questioned using different types of analysis for the

superstructure. You use the phasing, how can you apply the forces to the basemat.

So after some discussions the applicant revised the analysis method and for all three superstructures they used only equivalent static analysis method. So now you know the direction of the forces.

Of course in an earthquake they can oscillate in both directions so they have to consider the various phasing permutations.

And also involved is the type of combinations method when you combine three earthquake components there's two methods.

One is SRSS method and the other is the 100-40-40. And so KHNP spoke about the increase in the permutations up to 96. I don't know if you want to get into why it's so large.

So the combination of changing the analysis approach, increasing the number of combinations to 96 and what was the other thing.

And the last thing was these combination methods for directions are usually done at the member force level. So when you have a finite element in the basemat that's when you use a combination SRSS, 100-40-40.

Originally they had done it at the input level.

And the last thing is I believe they only did nonlinear analysis because the structure can partially uplift due to enough inertial load. But we questioned what happens if you don't have that uplift because they made some conservative assumptions.

So they did two cases with uplift and with no uplift. And then they enveloped those results.

So the cumulative effect of all these enveloping permutations you end up with a higher bearing pressure.

However, these higher bearing pressures were still shown to be less than the criteria they put in as a COL an information item.

MEMBER STETKAR: Thanks and I know that, the final conclusion. But thanks, that helps me a lot to understand. Thank you.

MEMBER REMPE: So I have a follow-on to my earlier question too about the swelling of the material used in the control rods.

Was it happening. Is someone here from the staff?

MR. ROCHE-RIVERA: Not in this panel.

MR. THOMAS: I think that was 3.6 I

believe.

MEMBER REMPE: It was RAI 928068 is all I can remember. I unfortunately am not allowed to have my -- I can't refer back to the actual documents.

MR. THOMAS: I heard a question this morning and I believe it was in the mechanical group.

MEMBER REMPE: Okay, so you need someone else here to answer that question.

MR. THOMAS: Yes.

MEMBER STETKAR: We're going to come to other parts of chapter 3 tomorrow I think is what they're saying. And it might be under those.

MR. THOMAS: Yes.

MEMBER REMPE: Okay. That sounds fine. Please ask them to let me know tomorrow.

MR. THOMAS: They will be there I'm sure.

MEMBER REMPE: Okay.

MR. THOMAS: If there are no other questions I guess we'll turn it over to the next presenters. Thanks.

CHAIRMAN BALLINGER: We're at 10 minutes after 10. We're again quite far ahead which is a good thing.

So what I'd like to do is take a 15-minute break and come back at 25 after 10.

(Whereupon, the above-entitled matter went off the record at 10:08 a.m. and resumed at 10:26 a.m.)

CHAIRMAN BALLINGER: We're ready to go. Bill.

MR. WARD: Good morning. My name is Bill Ward. I'm the lead project manager for the APR1400 review for NRC.

Can I check, David Heeszel, are you on the phone?

MR. HEESZEL: I'm here.

MR. WARD: Okay, thank you. So I'm going to present the APR1400 DCD chapter 2 site characteristics. This presentation will cover section 2.5. We presented the rest of these sections previously.

The review team are listed here and they are in the audience to answer questions.

In November of 2017 staff presented the advance safety evaluation report of other sections of chapter 2 except for section 2.5 which covers the geology, seismology and geotechnical engineering topics.

In the phase 3 subcommittee of September 2016 staff discussed the draft safety evaluation

report with open items for section 2.5 and we had no open items.

The section 2.5 staff coordinated with chapter 3 staff on crosscutting issues that originated from chapter 3 RAI 2558285 question 3.8.5-7. This was related to the settlement characteristics as they relate to the structural stability of the foundation.

And we stand by the previous conclusion that was presented to the subcommittee in September 2016, specifically that the applicant properly identified the appropriate geological, seismologic and geotechnical site parameters and properly identified the specific information to be provided as far as the future COL item.

About a month ago the staff completed the phase 4 advance safety evaluation report for section 2.5 and submitted it for review.

This report closed some of the confirmatory items we had previously identified and discussed with you in the prior ACRS meeting. It was based on revision 1 of the DCD.

We are now currently reviewing revision 2 of the DCD to ensure the confirmatory items identified in that report have been incorporated.

That's all I have for section 2.5.

MEMBER STETKAR: Bill, I had one item. I'm still puzzled a bit about something that I've been whining about through all of these subcommittee meetings.

And that is the assurance that the DCD contains enough information for the COL applicant to provide assurance that the interfaces between in particular the component cooling water piping tunnels and the auxiliary building and diesel generator building because they connect to those buildings can be designed and are acceptable.

Now I understand we've had several discussions about the essential service water building and the component cooling water heat exchanger building and those piping tunnels in the past.

And I know that there are now COL information items that specify explicitly that the COL applicant is responsible for the seismic -- the design of those buildings and the tunnels.

The only thing that I can't find is any specification in the DCD that tells the COL applicant that when they design those tunnels what's the allowable maximum differential settlement between those tunnels and the rest of those other buildings.

There's specifications in table 2.0-1 for

differential settlement among the nuclear island, the emergency diesel generator building and the diesel fuel oil tank building which are the three buildings in the scope of the DCD.

And there's a COL information item that says the COL applicant is responsible for designing everything else.

But it doesn't tell me how much differential settlement is allowed between the pipe tunnels and those other structures. Those all being seismic category 1 structures.

And it can be important if you've got big pipes running through there.

So I was curious how the staff, I understand how you've kind of resolved the COL responsibility for the structural design if you will of the SW and CCW heat exchanger buildings and the tunnels themselves. I'm worried about the interface.

MR. WARD: The discussion about the actual details of that, I know there's the settlement differences. I think that's more of a chapter 3 issue, but I'm going to ask --

MEMBER STETKAR: Well, it is, that's why. But the discussion about the settlement is actually in 2.5 as you've mentioned.

So right now it's why I've got chapter 3 and chapter 2 of the DCD open in front of me here.

MR. WARD: I'm going to ask if Vaughn or Ata want to start with that. Any discussion about the settlement differences between the tunnels and the other category 1 structures? How do we ensure that the COL has that proper information and ensure they don't exceed any.

MR. BRAVERMAN: Hello, this is Joe Braverman again.

The DCD does provide total settlements and tilt settlements I think in chapter 2 and certainly chapter 3 for other components like tunnels and piping which would be a COLA item.

In that case the COLA is going to have to come up with how much settlement the tunnel or buried piping may be subjected to and then assume worst case scenario out of phase. The building goes one way and the piping goes the other way.

And there are methodologies by civil engineers and standards on how to calculate --

MEMBER STETKAR: I understand all of that. It's in the DCD very clearly specifies maximum allowable differential settlement between and among three specific buildings.

It's 3.0 inches under static and a half inch under static and seismic loads.

What it doesn't do in the DCD, I can't find it. Maybe it's in there somewhere but I can't find it where the tunnels connect to those in scope buildings. I can't find anywhere where it says what's the maximum differential settlement at that interface.

MR. BRAVERMAN: Right at the interface.

MEMBER STETKAR: Yes, like where the piping would connect from the tunnel to the piping inside the building, for example.

MR. BRAVERMAN: When tunnels or piping connect to a structure they always put some kind of flexible joint.

MEMBER STETKAR: Would you allow 12 inches differential settlement?

MR. BRAVERMAN: No.

MEMBER STETKAR: Okay. Would you allow 0.0007 inches differential settlement? That's what I'm looking for so that when I as a COL applicant design my tunnels and my connections I know what margin I have to play with from the designer's perspective.

MR. BRAVERMAN: I believe the COL applicant would have to come up with the deformation

of the tunnel and then design a flexible joint that can accommodate it. And if it cannot then he has to design the proper tunnel to minimize the deformation so that the flexible joint can tolerate that deformation.

The -- KHNP cannot know ahead of time what a tunnel for certain site conditions, how much that tunnel may run. I believe the COL applicant would have to come up with that calculation and show that it can satisfy the needed deformation.

Should I try that another way if it's not clear?

MEMBER STETKAR: Well what leaves me hanging a bit is that in many cases the DCD makes rather explicit statements about certain criteria that a COLA applicant must satisfy to provide assurance that indeed the scope of whether it's a structural design, or whether it's a cooling water supply, or whether it's an offsite power supply, or whatever those things that are within the COLA scope of supply they're pretty explicit so that the COLA applicant understands what's expected when they do their site-specific design work.

And in most cases as I said whether it's an electric power supply or whether it's cooling water

capabilities or whatever they do tend to be fairly specific.

This is the only place and I've got it narrowed down because I'll give you the fact that they can build the other structures and make sure that they don't fall down as long as they connect up to the rest of the plant with reasonable assurance that the COLA applicant understands what margins they have to play with.

And I do get the notion that I can put some flexible hosing in there in principle and it'll take up the slop, but I don't think that we design seismic category 1 ultimate heat sink cooling water supplies with that amount of margin typically.

This is just sort of an aberration of the fact that they've drawn the line to separate the cooling water supplies from the rest of the stuff inside the nuclear island and the auxiliary building.

MR. WARD: I know the requirement for the applicant to provide this, I think it's item 25 the interface requirements.

And the review that's done, I don't know of a specific delta that is expected to be provided or normally provided.

MEMBER STETKAR: The only reason I bring

it up is I think this is the first time that we've seen this type of differentiation between the cooling water supply and the structural part of the cooling water supply and the rest of the scope of the certified design.

The details of the cooling water supply in terms of are you going to have cooling towers, is it going to be an open loop lake, that type of information is always left up to the COL applicant.

This I believe is the first one that we've seen where they've drawn the line and said everything that has to do with the essential service water and in particular component cooling water because that's the only thing that connects into the plant at the break point where those tunnels connect to the buildings everything outside of that break point is the responsibility of the COL applicant.

I think this is the first one we've seen. That's why I'm kind of ranting about it as I'll repeat.

If the staff feels, you know, obviously I'm only speaking as an individual. If the rest of the committee feels that there's enough collectively in chapter 2 and chapter 3 of the DCD and the staff's SER to provide assurance that those connections will

be designed and constructed adequately then it's fine.

MR. WARD: I think chapter 2 is primarily focused on soil characteristics and making sure that soil ranges that are defined are what will support the building that's being designed.

So I think this is all related to the interface requirements and what's in the structural.

MEMBER STETKAR: But I looked in chapter 3 and I don't find anything about the differential settlements. They're only -- at least in terms of the COL items.

MR. WARD: Okay. Well, I will take that back and we'll research it.

MEMBER STETKAR: Other than -- everything points back to that table 2.0-1.

MR. WARD: Right.

MEMBER STETKAR: But table 2.0-1 only has those three structures, the nuclear island, the auxiliary -- four structures. The auxiliary building, the emergency diesel generator building, the diesel fuel oil tank building. That's the only thing that's in that table.

MR. RODRIGUEZ: I just want to add. My name's Ricardo Rodriguez. I'm a geotechnical engineer. I worked chapter 2.5.

I want to reiterate what Joe Braverman said. The items that you see in table 2.0-1 are related to the structures that are within the scope of the DCD.

So talk at this point of structures that are meant to be built by the COL applicant I think we don't have that much information at this point to characterize the amount of flexibility or movement allowed for those tunnels we're talking about.

And I want to direct you to COL information item 3.7.6 which if I'm not mistaken tasks the COL applicant to perform seismic analysis of buried seismic cat 1, 5 and tunnels.

So I think at this point unless one of my colleagues want to reiterate a little bit more, but I think that's the position of the staff.

MR. WARD: What I will do is verify our guidance and regulation just to make sure that we're not expected to insist on any sort of particular delta. I'm not aware of that at the moment.

MEMBER STETKAR: Okay. Thank you.

MR. WARD: Any other questions?

CHAIRMAN BALLINGER: The good news is that we're way ahead. The bad news is that we're way ahead. And we have to do a little bit of adjusting.

I'm told -- 15 minutes ago I was told that the KHNP people for chapter 15 will be here in a half an hour. So that means that they're likely --

MR. SISK: This is Rob Sisk. Our chapter 15 people are not here yet.

CHAIRMAN BALLINGER: So I'm doing the math here. They should be here at about 11 o'clock. So we will recess until approximately --

MR. SISK: We do have the comment on the swelling but we're going to wait. I have the subject matter expert is here. He is on his way with the chapter 15. So you'll get it right from them.

MEMBER STETKAR: Rob, when do you expect them to be here realistically?

MR. SISK: We're going to check now. We got the same word you guys got.

MEMBER STETKAR: The problem is recessing and coming back at a specific time only to find out that they're not here is not all that efficient.

CHAIRMAN BALLINGER: I was going to define an uncertainty.

MR. SISK: We're doing a real quick check on the ETA now but of course with Washington traffic on the Beltway you never know. They're coming from Tyson's Corner.

CHAIRMAN BALLINGER: I think we'll recess and later in the vicinity at around noon. And we'll figure out where we're going then. Excuse me, 11.

(Whereupon, the above-entitled matter went off the record at 10:44 a.m. and resumed at 11:06 a.m.)

CHAIRMAN BALLINGER: Thank you very much for KHNP folks being able to accommodate us. I think you look like you're out of breath actually. It's all right. So, we can bring in oxygen if we need it. So, we're going to be back in session. And we'll start with Chapter 15. Thank you again.

MR. SISK: Okay. Thank you for the opportunity again. And at this point I'm going to turn it over to Mr. Woochong Chon to lead us through the Chapter 15 discussion.

MR. CHON: Good morning. My name is Woochong Chon. And let me think of something. There was heavy rains in D.C. area the day before yesterday. And I saw the news, and this one hotel was flooded. And all the guests was evicted at the very, early in the morning, like 2:30 a.m. 3:00 a.m.

And my colleagues and I were one of the members of that. So, the day before yesterday we took only less than two hours to sleep. And it was that

fresh experience for us. And also, it's a good time to think about safety.

If that hotel has a redundancy and independence concept, then that kind of things will not happen. But that hotel doesn't have any redundancy and independence like APR1400.

So, I'm sure that APR1400 plant has sufficient redundancy fact, and also independence. So, I'm here to present about the safety. And want to emphasize APR1400 is plenty of safety. Thank you.

Okay. Let's start of contents of this presentation. This presentation consist of three parts. First overview of Chapter 15 and summary of open items, and current status of Chapter 15.

And at the last page I describe the acronyms for your convenience. So, if you need those, some acronyms, then you can check the last page.

Okay. This slide shows overview of Chapter 15. Chapter 15 has a total of ten sections, especially 15.6.5 is LOCA analysis part. And other parts are related to transient analyst part.

This slide shows the list of submitted documents, and summary of RAIs for Chapter 15. First, in design control documents of Chapter 15, and non-LOCA, post-LOCA long term cooling, and Small Break

LOCA evaluation model. Those three items are technical report.

And the last line, the realistic evaluation methodology for Larger Break LOCA of APR1400 is supplemented as a tactical report. This tactical report is, became the base of Larger Break LOCA analysis of Chapter 15.

So here are the open items. Tech open items Phase 3 was raised. And those all open items are closed in Page 4. Okay. This slide shows the list of open items which are raised in Phase 3.

There are ten open items in Phase 3. And they are listed in this table. ADAMS Accession numbers are listed in the last column. And let's see some more details about each ADAMS.

The first open item in Chapter 15 is Periodic Reopening of the MCR Air Intakes, related to the RAI 7973 and 8470. KHNP identified that MCR ventilation system design provides for automatic selection of the least contaminated outside air intake.

And the closed MCR intake will re-open automatically, so that the intake with the higher radioactive can be selected again and closed. For this design staff required to show that the increase

in dose from both air intakes being open periodically can be encompassed by the current design margin, and design update is the PST.

As a resolution KHNP provided the additional information on the dose evaluation, including the dose increase can be compensated from the design margin of 20 percent.

The onsite ultimate dispersion factors of the control room have intake receptors, assuring the reduction factor of eight, instead of a factor of ten allowed by the Reg Guide 1.194. And DCD was revised.

MR. SCHULTZ: I have a question on the original basis of your selection of a reduction factor of eight. What I recollect is that you had decided to choose that reduction factor of eight before the questions came up from the staff as to whether this was going to be able to accommodate this issue.

What was the basis for choosing a reduction factor of eight, versus the reduction factor of ten? You could have not taken that approach. And I'm just wondering whether you had, what was the thinking behind using a reduction factor of eight, instead of ten, in the first place?

MR. CHON: Okay. Actually I'm not the expert for that area. So I, what I understand is just

give a 20 percent margin. The ten is defined in the Reg Guide 1.194. So, we want to set the 20 percent margin. So, conservatively we said eight instead of ten. But if you need a more detailed information I will ask to my experts and --

MR. SCHULTZ: I would like to know what aspect of the control room dose evaluation led you to a conclusion that you needed to have margin in this particular parameter. So, if you could ask that question, and provide a response on that I'd appreciate it.

MR. CHON: Okay.

MR. SCHULTZ: Not today. But --

MR. CHON: I will check.

MR. SCHULTZ: -- whenever you can. Thank you.

MR. CHON: Thank you. Okay. Next item, next slide. This slide shows the summary of open item related to the complete for our mixing model. In RAI 7917 NRC staff noted that the flow rate in Modes 4 and 5 when all RCPs are idle, and with one shutdown cooling train in service, may not be sufficient to assume complete RCS mixing.

Staff requested to provide the justification that the complete mixing model is

conservative, including any potential effect of incomplete lower plenum mixing. Also, they requested to demonstrate that the source range detectors can sense any postulated incomplete mixing.

As a resolution KHNP decided to add additional valve in the rear to make up water line to block the flow path from the unborated water source. In addition, KHNP added new Tech Spec 3.1.12 and its Bases to prohibit dilution operation.

And the quantitative analysis part for these Modes in DCD 15.4.6 was eliminated. And this item is also resolved.

MEMBER SKILLMAN: Let me ask this, please. How does isolating an unborated water source address the concern about mixing in the reactor vessel? Those two are absolutely independent phenomena.

MR. CHON: Please repeat one more time, please.

MEMBER SKILLMAN: How does isolating an unborated water source address the issue of mixing in the reactor vessel? I'll have to ask the staff this question. Because I, it seems to me that the question is really aimed at the idea of stratification, or thorough mixing in the very large reaction vessel, that's completely different than having a suction

source that's been isolated.

MR. SISK: Understand. I think we have a response here behind you. Andy.

MR. OH: Yes. This is Andy Oh, KHNP Washington Office. Originally this, the issue arose from during the shutdown mode for Modes 4 and 5.

If there's some, you know, fresh water came to the, came into the reactors, then it does not complete the mixing, then there's a reactivity problem can be happen, potentially.

So, in order to justify that complete mixing model we have to do some of analysis for, the CFP analysis, or some other thing. But it is very hard.

So, in order to reserve that complete mixing model we just installed the one valve to prevent of the source of the, you know, fresh water to come in, coming into the reactor.

So, after that is the complete mixing model problem. It's not a, can be some issue, potential issue. So, that's the way that KHNP responded to these issues to the staff.

More recently there's some issue for the mixing problem during raw mode. However, we just reserved this issue with some of the additional

modification of design. So, eliminate the whole source of the pressure water, potential pressure water came into the reactor during our raw model situation. Does that answer to your question?

MEMBER SKILLMAN: It answers a phenomenon question about density difference, colder water versus warmer water. The real driver for the reactor coolant system boron concentration is decay heat. It's natural circulation within the core, or within the loop, depending on who much decay heat is there, and what the concentrations are.

But it seems that the answer that's provided on the slide is pointing to a water source that can be colder, when in reality the question that the staff asked has more to do with how one ensures that the reactor coolant system is thoroughly mixed. And it seems to me that those are independent issues.

MR. SISK: This is Rob Sisk. I think, and I'll ask Andy to clarify if I got this wrong. I think the question is the fact that you're introducing unborated water into your thoroughly mixed borated water. And rather than deal with the unborated water being mixed in with the borated water, they've isolated it so that the unborated water isn't mixing in with the borated water.

MEMBER SKILLMAN: I can understand that logic based on the words that are here. But I would also suggest that at some point you're going to have to put in, if you're going to decrease the boron concentration, some over concentration water. And you're still going to have some problem like this.

MR. SISK: I understand that point. But that --

MEMBER STETKAR: Can I try something here. I haven't been following this. And I think it would be useful when the staff comes up for them to elaborate on their initial concern here.

What I see this particular modification responding to is an issue that was identified first by the French, when they did shutdown risk assessment and identified a condition where you could fill a loop with pure water, and then start a reactor coolant and inject the pure water into the core, resulting in re-criticality.

It's not the boron mix, it's boron mixing in the sense that if the pump was running you would get mixing of that. It's my understanding that this valve, and the corresponding tech spec protects against that phenomenon.

Is that correct, Andy? And again, I'd

like to hear from the staff what their initial concern was. Because we may be talking about two different --

MEMBER SKILLMAN: Phenomena.

MEMBER STETKAR: -- phenomena of two different so called mixing phenomena.

MR. OH: Yes. And you can have two kind of the mixing problem. My memory tells that, yes, that's right. And a slug.

MEMBER STETKAR: A slug.

MR. OH: It's slug. And then it's when RCP is re-operated, that slug is, goes into the --

MEMBER STETKAR: Yes.

MR. OH: -- documents and records. That's the one, another thing is also, yes, this one for pressure water source. It's just in. So, two kind of things that combine together.

MEMBER STETKAR: This also protects you against kind of a gradual boron dilution.

MR. OH: So originally I think for this RAI deal with two kind of things. But we drove the first two things in a slug, with a different way. And the second thing is we also isolated the source of the unborated water using a just isolation valve. But the title is only for the complex completed mixing model. But inside of here it deals with the two issues.

MR. CHON: We have another HRS meeting for long term core cooling part. And the slug with unborated water flow into the core. That RAI will be treated in the long term core cooling HRS meeting.

Okay. This slide shows the summary of open item related to rate of trip during boron dilution at power. In RAI 8668 NRC staff noted that a slow reactivity insertion transient such as boron dilution at power would not be protected by the variable overpower trip.

Staff requested to provide which reactor trip would occur for boron dilution event at power. As a resolution KHNP provided the clarification that the core protection calculator system, DNBR trip or CPCS auxiliary trip can protect the DNBR limit during a slow reactivity insertion event. And this item is also resolved.

And next open item shown in this slide is Small Break LOCA Dense Break Spectrum Analysis, related to RAI 8337. Staff requested to provide the result of finer break spectrum for both the DVI line break pumper discharge rate break.

KHNP performed the dense break spectrum analysis for the DVI line and pump discharge rate break. Break spectrum analysis number is increased

from 10 to 39. Here 37 is missed, typo number. So that now break spectrum analysis number is 39.

The limiting DVI line break size is changed from 0.4 feet square to 0.1364 feet square. The LOCA criteria is still satisfied. And KHNP revised the RAI response in technical report, and also DCD Chapter 15.6.5. And this item is resolved.

MR. SCHULTZ: Can you provide some indication, and I don't have the table in front of me, but as to what margin you had before you did the new break spectrum analysis? And what margin you had after the fine detailed break spectrum, and changed the bread size?

MR. CHON: You mean margin for the --

MR. OH: What you say --

MR. CHON: -- temperature?

MR. SCHULTZ: Yes.

MR. CHON: In previous version we have four DVI line break and four pump discharge rate break, and also one IOQSLV event. And the other one is instrument tubal break. So total nine break size was evaluated.

And new methodology we, DVI line break is a total 15, and pump discharge rate break is 17. And also IOQSLV event is one. And instrument tubal break

is one. And more smaller size break is five cases.

MR. SCHULTZ: No.

MR. CHON: The detailed PC number will be answered my colleague, Mr. Lew. The PC number for previous one and new calculation one.

MR. SCHULTZ: To a microphone, please. Thank you.

MR. LEW: My is Kaeyeol Lew from KHNP. So, previous limiting PCT is 1,166 every 39. And new one for PCT is 1,684.

MR. SCHULTZ: So, it was a good thing to do the more detailed break spectrum analysis. You did get substantially different results. So, that was a good thing to do.

Just, I wanted to get this on the record that the fine detail of a break spectrum analysis can be important in the analysis result that's derived. Thank you.

MR. CHON: Okay. The next open item is upper bound on the Small Break LOCA break size. This open item is related to the RAI 8488. The staff requested to demonstrate that the 0.5 feet square break size remains the largest Small Break LOCA which must be considered.

KHNP explained that the analysis with the

breaks larger than 0.5 feet square are treated by the current Large Break LOCA methodology. And KHNP also presented the result comparing the Larger Break LOCA PCTs.

The PCT calculated by the Larger Break LOCA was greater than the PCT obtained from the Small Break LOCA methodology. It also come from that both the Larger Break LOCA and Small Break LOCA PCTs are much less than 2,200 Fahrenheit. Hence, this item is also resolved.

MEMBER REMPE: So, I have a question. In, it pertains to questions I'll have later on with, when we talk about the Large Break LOCA methodology. But your calculations were done with RELAP/MOD3.3/K, right?

MR. CHON: Correct.

MEMBER REMPE: And I don't know if I asked this when we discussed it previously, but this is a version of RELAP that comes through ISL? Or is you version that you obtained from some organization in the U.S., right?

MR. CHON: We received the RELAP5 version from the NRC.

MEMBER REMPE: From the NRC?

MR. CHON: Correct.

MEMBER REMPE: Okay. So, NRC relies on an organization, ISL. And as with the many codes there's always errors that are found in future years. So, basically you are maintaining that relationship with NRC and ISL to incorporate error improvements, corrections periodically?

MR. CHON: Yes. Well, there is one organization, which is KAMP.

MEMBER REMPE: Right.

MR. CHON: We also one of the member of KAMP. So, we are receiving the error informations periodically. But this RELAP5/3.3 applied in this calculation is, error correction is not applied.

MEMBER REMPE: Well, probably when you obtained it, it had been updated to provide those error corrections. But then, in the future you'll be monitoring error corrections that come, and assess whether you need to do anything, is what I'm trying to get to.

MR. CHON: Yes. We are doing assessment --

MEMBER REMPE: Okay.

MR. CHON: -- every time.

MEMBER REMPE: Thank you.

MR. SCHULTZ: I have a question on this

slide. The question was to demonstrate that this break size is the largest small break which must be considered. And so, you did a comparison between the large break methodology and the small break methodology.

Why is it that you want to conclude that the large break methodology is greater than the PCT for the small break? What was it that told you that you were therefore done with the analysis to answer this question?

MR. CHON: Okay. From the smallest break size, from the 0.5 it is clear we are applying small break methodology, with C plus 4 AS code. And from the 0.5 to the 100 percent break size is, will be handled by Larger Break LOCA with RELAP5. And that's the ultimate methodology.

So, even, excuse me. The PCTs at the 0.5 feet square, the Larger Break LOCA PCTs are higher than Small Break LOCA. And these PCT will be compared with other total range of Larger Break size ranges. And we will select the highest PCT, including the 0.5 PCT.

MR. SCHULTZ: So, that's why it's good. It bounds the small break result at that location, for that break size. You wouldn't feel more comfortable

if they were the same PCT?

MR. CHON: Yes.

MR. SCHULTZ: But you wouldn't feel comfortable, as comfortable if the large break analysis methodology gave you a lower number than small break. Is that, am I right with that?

MR. CHON: In Small Break LOCA Permanent and Larger Break LOCA Permanent is a little different. So, C plus 4 AS code is adopted for the Small Break LOCA Permanent.

So, even Larger Break LOCA will apply could calculate the higher PCT. But still we personally delete the Small Break LOCA code. It will calculate the Small Break LOCA PCTs until 0.5 feet square break size.

MR. SCHULTZ: Okay. Thank you.

MR. CHON: Okay. Let's move on to the next item. And this slide shows the summary of open item related to the impact of thermal conductivity degradation, which is TCD.

In RAI 7954 NRC staff noted that the FATES3B code does not account for the effect of TCD, and also requested a TCD impact on the fuel rod design and safety analysis.

FATES3B code is used for the fuel rod

performance analysis, and generation of interface data for safety analysis. To resolve this open item TCD penalty was composed to mind based on analysis results compared with the experimental data at various burnup.

KHNP performed the analyses for the design evaluation for the related safety area with TCD penalty. For CEA ejection, peak radial average fuel enthalpy is increased less than 11 calorie per gram, which is much lower than 230 calorie per gram.

And peak centerline temperature is increased about 317 Fahrenheit at HFP case. And no fuel centerline melting is occurred. For Larger Break LOCA part the details of TCD effect will be discussed in the Larger Break K/REM topical report section later.

And DCD Tier 2 chapters and Larger Break LOCA topical report were revised. The RAI, including impact of TCD, has been resolved.

And this slide shows four open items, which are loop seal clearing and reformation, and post LOCA boron precipitation, LOCA deposition model, and boron dilution. The details about these four items will be treated in the long term core cooling section separately.

Lastly, this slide summarizes the current

status of DCD Chapter 15. Chapter 15 is completed. KHNP continue to monitor Chapter 15 to assure any conforming changes are addressed.

Ten open items. Those were identified in Phase 3, has been resolved with adequate and sufficient discussion with the staff. Changes in Chapter 15 as reviewed, and marked up in response to the RAIs are incorporated into the Revision 2 of the DCD.

Thank you so much for listening. And I will happy to answer the question you might have.

CHAIRMAN BALLINGER: Questions? Thank you.

MR. CHON: Thank you.

CHAIRMAN BALLINGER: Is the staff ready to go?

(Off microphone comments)

MR. STECKEL: Hi, good morning. My name is Jim Steckel. That's Michelle Hart on that end and Shanlai Lu.

We'll do our staff's presentation on our review of Chapter 15. And we'll go through some of the items that were discussed as far as RAIs and their close out.

We're going to start with Michelle and

we'll begin with Slide Number 2. Michelle.

MS. HART: Thank you. I'm Michelle Hart, I'm in the radiation protection and accident consequences branch and I reviewed the design basis accident dose analyses in Chapter 15.

We did have one open item in the last phase of the review for the APR1400, and it was the modeling of the control room emergency makeup air cleaning system operation and the design basis accident dose analyses, the control room compatibility analyses.

The design certification did not clearly describe the system operation where the control logic automatically re-opens the close air intake. They have two diverse air intakes at a preset interval for a short period of time to determine which air intake has the most contamination and then re-closed that contaminated air intake. So they always have the lowest contamination being taken into the control room.

The dose analyses did not make that clear that that was the operation, and neither did the discussion in Chapter 9 or Chapter 6. So we asked some questions about that.

The response, the REI responses for that

describe the modeling assumptions for the damper operation. And that they did make revisions to the design certification, Sections 9.4.1 and 15.0.3, to more clearly describe the system operation and the dose analyses assumptions that the interval between the openings was an hour and the duration of the openings was one minute in their assumption.

And that assumption was accommodated by the 20 percent margin that they had from the X/Q assumptions that they had for the intake.

They also revised their ITAAC to clarify the damper operation, and that will be discussed in Chapter 14. That's really where the question came from was Chapter 14 and the ITAAC.

And they revised COL Item 9.4(2) and they added new COL Item 15.0(2) and (3) to identify that the COL applicant verified the damper re-opening interval and duration are the same or are accommodated by the current dose analyses. So that they're aware of those.

The assumptions on the interval between the openings and the duration of the openings. And so that did resolve the issue.

MR. SCHULTZ: Michelle, I wanted to come back to the margin that was used. And I wanted to be

sure that there's enough information in the final document to describe to the next evaluator of dose to, not double take on the margin --

MS. HART: Right.

MR. SCHULTZ: -- because it, that's really one of the things I'm, or that's really the main thing I'm concerned about.

It was roughly stated that, well, this will be accommodated by the 20 percent margin but it, I know it's within. You've given the parameters and you wouldn't expect a very large delta associated with that, but you've used up some of the margin in that 20 percent value, and that needs to be taken into consideration when the dose analysis is done.

MS. HART: Sure. That was certainly one of our concerns as well, that they were using up some of that margin and it was not clear that the margin was being used.

And so one of the effects of the COL items that were added in Chapter 15 is to ensure that the COL applicant looks at that analysis and determines if they're, you know, to make them aware that that margin is being used to accommodate these things.

MR. SCHULTZ: That's what I wanted to be sure was done.

MS. HART: Yes.

MR. SCHULTZ: Because I could see that being taken into account, perhaps in the uncertainty of the X/Q value --

MS. HART: Some other --

MR. SCHULTZ: -- sometime in the future.

MS. HART: Right.

MR. SCHULTZ: And they only have part of that left --

MS. HART: That is correct.

MR. SCHULTZ: -- based upon your discussions with the applicant.

MS. HART: That is correct.

MR. SCHULTZ: Thank you.

MEMBER SKILLMAN: Michelle, as you described, or you presented the question and how far you had to go to get the answer, was there a, is there a description of the quality level of the control system that makes the selection for which air intake is ultimately chosen?

Is that a nuclear safety related or is that QA1, was there exploration to that level, to the hardware, so that there is confidence that the selected air intake is the air intake that is truly desired and there is confidence that that equipment

will do what it's supposed to do?

MS. HART: We did look into that. I have to admit that I don't remember what we determined.

I am not the reviewer for that section. It would be something that was also evaluated under Chapter 7, in general. But we did look into the quality of the selection.

MEMBER SKILLMAN: Okay, thank you.

MS. HART: Any more questions? I will turn it over to Shanlai Lu then.

MR. LU: Okay. All right, Shanlai Lu from staff, reactor systems. So I'm going to go over the remaining open items and how we closed them as part of this phase of the review, okay.

So, I'm going to go item by item. At this point, all ten open items have been closed. Some include the non-term cooling aspect. I also include my presentation as the high-level summary, for example, boron precipitation, boron dilution in theory.

But the details can be, if the committee wants to know the more details, that will be in June's presentation. But today, I already include them as the high-level summary.

Okay, so I'm going to go through item-by-

item. The first item is related to fuel pellet thermal conductivity degradation, the TCD issue.

The issue came from Chapter 40 review. 4.2 fuel design and then Class 7 KEPCO. And the BRD presented the reasons why we had this issue, why we asked for these RAIs, additional information.

And the reason, really the FATES3B, the old, ancient code. But they have been used in that design basis for this run of the, design certification analysis, and in which most of the thermal conductivity degradation phenomenon.

And with that, we performed a confirmatory analysis, we found because of that it may miss the centerline temperature at full power, several hundred degree Fahrenheit. That's a lot.

So therefore, we launched an RAI. Not only in Chapter 4.2 but also as the relevant Chapter 15 analysis, we asked for what's the impact.

So, as part of a few topical reports in the 4.2 presentation to you guys, we already resolved and presented to you, once the Staff considers the, a profit conservative penalty. We add it on top of the FATES3B, and analysis package can go forward.

And now we're going to talk about the cost of the penalty. So based on the Staff's, the penalty

on the centerline temperature and on the FATES3B computer code. And they performed additional analysis anywhere impacted by the initial fuel temperature, centerline temperature calculation, for the Chapter 15 part.

So this part actually is close but I am going to go through that a little bit, okay. The CESEC code is an old code really, to calculate the ATWS code and the ATWS scenario and then the system response and the power.

And one part is the impact that could be, by the initial temperature, was the gap conductance. And they found, they already choose the code, already choose very conservative of gap conductance for that particular purposes.

And it was not intended to capture the TCD but they already got it. So, with the additional temperature increase, due to the penalty and then the impact on the gap conductance is minimal, so it's not an issue for that one.

And that there is also, we also have the, we also questioned them, what's the feedback. And then you have kinetics and then you have the operating factor, feedback coefficients might be impact because of your fuel temperature is elevated by several

hundred degrees. That is a hot thing, hot spot.

And they planned, they provided additional analysis evaluation and they found the, when they select the operating feedback coefficients, the upper bound value and then conservative of the already existing analysis, already found this temperature delta.

So therefore -- yes, Dr. Schultz?

MR. SCHULTZ: So each of these were separate evaluations? Bullets 1 and Bullet 2 there.

MR. LU: Right. And then they combine the value to the CESEC code, the analysis existing, and see whether from those two perspectives, are they really still credible to produce conservative results. And then our conclusion, based on what we provided, we think there is sufficient margin error.

MR. SCHULTZ: You evaluated the uncertainty that was --

MR. LU: That's right.

MR. SCHULTZ: -- provided as well as the gap conductance values --

MR. LU: That's right.

MR. SCHULTZ: -- and determined that they were appropriate?

MR. LU: That's right.

MR. SCHULTZ: They're best estimate, or whatever they call their values --

MR. LU: No, no, I'm sorry --

MR. SCHULTZ: -- were appropriate?

MR. LU: Yes.

MR. SCHULTZ: The conservatism and then the delta.

MR. LU: Yes, that's right. That's part of the RAI responses they provide to us.

MR. SCHULTZ: Thank you.

MR. LU: Okay, next slide. Okay, then thermal codes, other codes also impact the CETOP thermal margin analysis.

And of course we, because of the initial temperature, fuel centerline temperature could go higher than what we calculated, we wanted to know what's the impact on the DNBR margin during the transit.

And then we found out, oops, and then that part of the code is not sensitive, even to this issue. So CETOP margin, thermal margin analysis is not an issue.

We did see the rod ejection case and it can be, you know, changed. And then because of that one, so they performed additional new analysis based

on still the existing code. The code is not an impact of the, it's applicability to analyze this scenario.

But because of the pellet thermal conductivity multiplier applied in the STRIKIN-II is to be increased, to handle the penalty we added on top of FATES3B, so we re-run code. And then the good news is, there is no cure in the fuel failure centerline.

Centerline temperatures did, the prediction did go high. Higher than what previously calculated. But it's still much lower than the failure criteria there.

So therefore, the current Chapter 15 4.8 with the revised analysis, they address the Staff's concern about thermal conductivity degradation after we applied the penalty.

Okay, as the KHNP mentioned, they also updated their Large Break LOCA topical report to take into account how that methodology can take on the additional penalty. So that will be presented after lunch probably.

So with that, anything related to TCD on Chapter 15 we did provide additional analysis and RAI as part of the official RAI responses on the docket and we, Staff, reviewed, actually we performed quite a lot, confirmed to analysis based on our site, and we

don't see a challenge for the design. So therefore, yes.

MR. SCHULTZ: Just a comment really.

MR. LU: Sure.

MR. SCHULTZ: Unfortunately, you've had a lot of experience applying penalties associated with the thermal conductivity degradation through analyses --

MR. LU: Right.

MR. SCHULTZ: -- and so, using that experience, have you been able to determine that the application for this Applicant, had been done appropriately, to demonstrate that even with penalties, the analyses meet the criteria that they're designed to meet.

At the same time, I'm sure you would have preferred if the application had used methodologies that incorporated the thermal conductivity degradation. And I would certainly hope that in the future, that those analysis methodologies are developed.

Not only including safety analysis but also physics analysis and other applications where thermal conductivity is important. Thank you.

MR. LU: Yes, I think that's, we're on the

same page. And that's the reason, when we had to pre-submit a QE audit, we already identified the issue.

But we knew that to redo that part of the review and then also for them to integrate a brand new code into this fabric of entire suite of codes, it's almost a type audit. It may not be workable for that.

And by the same time, they already told us, they already submitted their code. Look for code. I think first is they asked for it on the field.

And then that part already addressed the TCD issue. But they send through the review process of the Korean regulator. And then that part was recently approved. And it was used for Shin Kori Unit 3 operations, right? Yes.

So I think that's what we, we actually interacted with teams on this particular issue to end that. And that we wanted to know, do these really say additional challenge after they use the code, which has already addressed the TCD.

From the design perspective, there is no challenge of the, in the original design criteria. The markings are reduced but it's still past their margin there.

MR. SCHULTZ: That's good, Shanlai, I'm glad that you put that onto the record today because

we've talked about this in general on how you got some specific advances that they have done --

MR. LU: Right.

MR. SCHULTZ: -- that I'm glad to hear about. Thank you.

MR. LU: Yes. I think this was based on your comments last year, so we went out and talked through them, talked to them, and find out what, how they resolved this issue.

And the Korean regulator came, they told us how they resolved with KHMP over the Korean application. I think going forward, and they come back with really COL, I think that they may have better methodology to go forward.

MR. SCHULTZ: Okay, thank you.

MEMBER REMPE: So, before we leave that topic --

MR. LU: Yes.

MEMBER REMPE: -- and we may discuss this more in the Large Break LOCA methodology, but it was observed in some of the material I read, that RELAP 3.1 versus 3.3 uses a different thermal conductivity degradation model.

MR. LU: Okay.

MEMBER REMPE: And that the 3.3 model is

less conservative. And this is an NRC code that ISL is helping you maintain and why was that done?

MR. LU: Okay. I mean, I think the best place to address that question is the presentation later, but I can add on top of it. It's irrelevant to the different RELAP versions.

The penalty we added on top of that is on the FATES3B, how you initialize it. So how do you capture the initial temperature distribution, the initial sort of, you now, as the initiate steady state condition. That's the major contributor to the initiate stored energy there.

And definitely, you are right. And then there is the RELAP 5 versions of different TCD model waiting the different version. That will be covered by today, this afternoon's presentation.

MEMBER REMPE: Okay, thank you.

MR. LU: Okay. All right, let's move forward. There is one more slide.

All right, we just talked about this. So, we did identify two open items, 15.6.5-1 and 2. And then basically capture the Large Break LOCA topical report needs to be reviewed.

And we are in the process to wrap it up, the SER. And you can hear this afternoon

presentation. We are proving that topical report.

And the analysis results for 15.6.5-1 LOCA analysis. So that part is closed right now. Both open items are closed going forward.

All right, inadvertent boron dilution during Modes 4 and 5. Okay, I think I answered Mr. Skillman's, Gordon, your question. I think that's the page.

The phenomenon, okay, you asked a question to KHNP, I think that we already addressed that one. From a Staff perspective, let me give you the, where we came from.

Okay. So during Mode 4 and 5 there is an inadvertent boron dilution due to the actuation of freshwater injection. It can happen. It can happen.

So, you know, the successful criteria or the safety concern here is, are you going to reach critical. During full Mode 4 and 5 it's not pleasant. You have the core going critical during Mode 4 and 5, that has to be stopped.

So, the question becomes, when you really have inadvertent freshwater injections through the freshwater source, as CVCS injects there and less than half critical.

So, their initial analysis assumed, when

there is a slug off of freshwater coming in, the freshwater pump injects into the RCS system, the water were immediately diluted and mixed with the entire RCS coolant.

So previously, it's all-natural circulation but it's borated water. And then you have decay heat, and it depends on Mode 4 and 5, initial starting point.

If it started from, started from zero power come down, you don't have a whole lot of decay. But if you have full power come down to Mode 4 and 5, you still have a lot of decay heat to remove. You have natural circulation going on.

But the situation becomes, we checked into their analysis to say, they took the credit of these analysis assuming complete mixing of the freshwater, immediately after water is injecting into the RCS. So the question, where is this, is this conserved. That's the reason we asked that question.

So, when we asked the question then we found out, and then based on the interaction with them, they said it may not considered. But it's really extremely hard to, within the time frame, to solve this problem automatically.

If it's not solved automatically, then

we'll have to review their CFD codes of the entire RCS system. Which, you don't really see many cases using CFD codes for that case.

So therefore, then the bounding analysis is performed, indicating there is a potential possibility the core could reach critical.

Depends on how you assume that a slug of water coming into the RCS might lead down into the downcomer and the mix of the, because it's colder than the system, it's colder water, and the freshwater comes in and it may not have mixed very well, it will mix, but it might not mix completely with the entire RCS, to give you a number at the core inlet, what's the boron concentration. So that's the reason we asked the question.

So this also, the only caution within the reactor system side, actually, it does require additional significant change. Okay, that's the part.

And then based on, not only just because of the difficulty of the analysis to performed CFD, but if you perform a bounding analysis of assuming non-mixing, the freshwater will reach the inlet of the core, you reach critical. So therefore, the safest way to do that, is not perform analysis to take the credit, which may not be available during the time

frame of this particular review, but is to just secure the freshwater source.

Which is, they added a new isolation valve in the reactor, makeup a water line to block the flow paths. But following certain tech specs back.

It can, this can still add freshwater, it just use that one to control and make sure that part is isolated. And then the operator can check, during a Mode 4 and a 5, this valve is closed. That's the reason.

And then the, related to that valve, and then of course there is additional tech specs changed. And then they introduce additional tech specs change.

With that one, we do not see any potential concern, any more, about the criticality due to the freshwater.

MR. SCHULTZ: Thank you for the explanation. And I see that now in the RAI.

MR. LU: Right.

MR. SCHULTZ: As, John pointed out --

MR. LU: Yes.

MR. SCHULTZ: -- this was a concern about a slug, and it really gets down to, where are you when you start an idle pump.

MR. LU: Yes.

MR. SCHULTZ: Or the flip side is, when no pumps are running.

MR. LU: That's right.

MEMBER STETKAR: Just for clarity, the text spec requires this valve to be closed in Modes 4 and 5 --

MR. SCHULTZ: 4 and 5.

MEMBER STETKAR: -- when all RCPs are idle. So when you have no force circulation from the reactor coolant pump.

MR. LU: Yes.

MEMBER STETKAR: So, it allows you to start up a reactor coolant pump so that you can eventually start up the plant and start to dilute.

MR. LU: That's right. The slug issue by itself, we are, actually, I have another point related to that boron dilution item. It's within Small Break LOCA, you have a loop seal, you have a slug.

But this particular, if you have freshwater injection, it may form a slug too. But RCP may not be activated to press the water through to the core.

So therefore, the issue really, it's slow mixing transient, can that be causing any Staff's concern, a safety concern of potential re-criticality.

MR. SCHULTZ: Thank you, Shanlai, for the explanation.

MR. LU: Okay.

MR. SCHULTZ: I understand.

MR. LU: Okay.

MEMBER MARCH-LEUBA: I wanted to make a comment. You were kind of apologizing --

MR. LU: Yes.

MEMBER MARCH-LEUBA: -- that there was not enough time to run a sophisticated analysis to demonstrate that the problem even exists.

MR. LU: Right.

MEMBER MARCH-LEUBA: I think that the solution that I came up with is better than the analysis. Because you actually --

(Laughter)

(Off microphone comment)

MEMBER MARCH-LEUBA: Yes. You prevented the accident and you improved the safety of the reactor.

MR. LU: Right.

MEMBER MARCH-LEUBA: And I wanted to congratulate you for having done that, because, all the applicants would have dug their heels into the sand and push backed and say, it's going to happen,

let me analyze it today until you agree with it. They just got rid of the event and perfect, improve the safety of the reactor.

MR. LU: Yes, I think I'm one hundred percent on that on.

MEMBER SKILLMAN: We have operating experience where loops can be stagnated and independent from each other. And so the idea of a freshwater slug is very real.

MR. LU: Right.

MEMBER SKILLMAN: I mean, this is not a myth, this can really happen.

MR. LU: Yes.

MEMBER SKILLMAN: So, I commend you for pursuing this and them for making the change. I understand that it was just the way the wording was in that question and response that caused me to say, they're really two phenomena.

You do have natural circulation if you have decay heat being generated. But the real issue is the isolation from a loop.

Where you can get that isolation in cold water, you can in fact have a completely barriered slug that the rest of the loops, the other three loops, do not know is there.

MR. LU: Yes, that's right.

MEMBER SKILLMAN: You don't find it until you start that pump.

MR. LU: Yes.

MEMBER SKILLMAN: Okay.

MR. LU: Yes, actually the staff working on this, the issue, was first right away over there. And he did this job.

MEMBER SKILLMAN: Okay.

MR. LU: That was Jeff and --

MEMBER SKILLMAN: Yes, that's good. Thank you.

MR. LU: So I am just presenting on his behalf, but he did a good job there. And also, in the KHNP, they realized the analysis, the balance between the hardware change and the analysis. And I think this issue was closed.

Okay, next one. Okay, variable overpower trip during boron dilution at power. And also, its potential boron dilution event but it's a standard dilution event.

And due to the CVCS injection there, what we really want to know at that time was not really a safety concern, we just wanted to know, do we have a variable overpower trip which will be activated or not

during this slow transient insertion. And, again, slow reactivity insertion transient.

And then, they actually address this issue because they pointed out that during this type of event, because the core protection calculator system actually intended to protect the DNBR. And then based on the power level and the, would capture that.

So therefore, for this particular event and the variable overpower trip signal, whatever, actually comes in much later than the core calculator power trips.

Therefore, and based on Staff's review on Section 4.4, thermal hydraulic event in CPCS, and we felt that that's already been protected, there is no need to have variable overpower trip to actuate. So therefore, this issue is resolved based on their clarification.

Next slide. All right, we just talked about, I think KHNP also presented this one. The upper bound break size and PCT, and also, the break spectrum.

The reason, when we started to take a look at the Small Break LOCA and our reviewers, Syed Haider, Syed Haider was there. And he reviewed this, the Small Break LOCA, and he found that, okay, the

break spectrum is too course, the break size.

And our caution, based on our confirmatory analysis, we're not so sure they're capturing the most limiting break size, in the Small Break LOCA spectrum. So we asked that question.

And that's one of the questions. Another question is, between the Large Break LOCA topical report analysis and the Small Break LOCA, do they bound the entire spectrum, including the junction point between the Small Break LOCA and the Large Break LOCA?

And they did as part of responses, they indicated, based on both Large Break LOCA and the Small Break LOCA analyses, the entire spectrum, starting from very tiny to all the way to double and break, that's everything calculated, PCTs less than 2,200, we're fine.

And then the much denser spectrum, break spectrum analysis, pointed out higher PCT. However, still less than, much less than the 2,200, therefore this issue is closed.

MEMBER MARCH-LEUBA: I've read so many things that I forget. I know the staff has performed confirmatory analysis for the large break.

MR. LU: Right.

MEMBER MARCH-LEUBA: Do you do the small break confirmatories?

MR. LU: Yes, we did. We asked them to run. We also ran our case, too. And I think one year ago -- I don't know whether you were here one year ago? Okay. Then, you maybe were not here.

I think we present both small BLOCA and large BLOCA and this afternoon the research where we are also presenting TRACE analysis in support of Large BLOCA topical reports.

MEMBER MARCH-LEUBA: And does it agree more or less on the break size? Because the break size move from -- it became smaller. It's not the trend we see with this?

MR. LU: Maybe I missed the question. I'm sorry.

MR. SCHULTZ: The break size for a small break.

MR. LU: Oh, okay.

MR. SCHULTZ: Has it changed?

MEMBER MARCH-LEUBA: It's difficult. The worst break size.

MR. LU: Oh, yes, yes, yes.

MEMBER MARCH-LEUBA: When it went through the denser.

MR. LU: Oh, yes. It moves. It moves from the previous -- when they had a much coarser break spectrum, which is about 10 cases, I remember, right?

MEMBER MARCH-LEUBA: Yes.

MR. LU: Now you are running the 39 cases. It's a big effort there because, you know, they have to redo a lot of work. And then when they performed this one, they found out after addressing all those comments, issues, in particular for the break sizes spectrum, after they added additional 239 cases, you know, that does have a shifting peak cladding case versus we applied the curve off of a cladding tempered PCT break spectrum, that break spectrum changes. If that's the question, you're asking, yes. That's the case.

MR. SCHULTZ: Did the staff calculation have a comparative break size to what they did for small break? About the same range?

MR. LU: Okay. I cannot remember exactly. I think we got comparable results with their RELAP5 calculation. But their small break load commenced with our foremost limiting case. It's based on very conservative Appendix K model.

So, therefore, I cannot say exactly

whether they are identical. They are not identical anyhow. But which one predicts higher or lower, I cannot remember now. But I can pull that out if that's something of interest to you.

MR. SCHULTZ: But just with regard to this issue --

MR. LU: Yes.

MR. SCHULTZ: You have a large break peak and you have a small break peak.

MR. LU: Yes. That's right. That's right. Actually --

MR. SCHULTZ: So that's why this crossover is okay.

MR. LU: Yes. That's right. That's the reason we have this case. All right.

MR. SCHULTZ: Are we good?

MR. HAIDER: Hi. This is Syed Haider. I was the reviewer for the small break LOCA. I would like to make one more clarification here that the peak of CAREM PCT that was found by the applicant using their methodology by the loop 1684 degree Fahrenheit while the TRACE calculations that the staff performed was for the tune of both 1200 degree Fahrenheit. I don't have the exact numbers, but the applicant's PCT was much larger than what the staff found.

MEMBER MARCH-LEUBA: And you have given that to the conservative assumptions?

MR. HAIDER: Yes. That's precisely right. The conservative assumptions in their S1M methodology.

MEMBER MARCH-LEUBA: In that point, do you find through pricing that this low break LOCA is less conservative or gives you a lower number than the large break LOCA of .5 feet per square? You know, what they told us is large break gives you a higher PCT than these low break methodologies. And you're telling me this low break is very conservative.

MR. HAIDER: Actually one area was about doing detailed analysis on .5 in a 1 foot square, and we did find that the large break at 1 foot at the intersection was more conservative. That's what we found.

MEMBER MARCH-LEUBA: Okay.

MR. LU: All right. So after all I think it's way below 2200. There's no delineage or the closest item. The next item.

Boron precipitation during long-term cooling actuation because of staff RAIs, they re-performed the analysis of boron precipitation during long-term cooling.

So details I can present in June if you

guys want to hear more, but the summary is here. And this did introduce addition of kelvin change, but it's good news is just the EO piece.

So the time of switch-over from DVI line to hot leg. And initially their analysis indicated that they can survive for three hours with that switch-over.

So after requesting their methodology of how they really assumed the mixing in the core and upper part of the plenum and then the revisor analysis, they found, okay. There is a chance that we have to -- they may get into the critical situation during LOCA scenario during long-term.

Normally, we do not allow critical. And then so because of that perspective, and they said, all right, we are going to perform a conservative analysis and then to control the switch-over time from -- to reduce the switch over time from three hours to two hours. And that issue was resolved.

MEMBER MARCH-LEUBA: And while I like this type of modification to make the reactor safer, you also have to evaluate it doesn't produce unintended consequences.

MR. LU: Yes, sir.

MEMBER MARCH-LEUBA: Did you guys look at

-- I mean, does it cause any problems?

MR. LU: You're right. I agree with you. I think that from the perspective, you know, if one party push too hard and maybe another part pops out.

And in the situation for this particular one, I think it's kind of isolated at this point what we believe that it may not cause too much heartburn for operator to take action earlier from three hours to two hours to switch-over. And even before, you know, they have to switch from three hours to half hour, then we may have problem. You're right.

But from three hours to two hours, we don't consider it's a challenge. And they should be able to accommodate by their EOPs. Okay.

Next item. The burn dilution during long-term cooling phase. And then I think KHNP indicated they plan to give you a more detailed presentation in June. We can talk more, too. Here's a summary.

And then there are two parts up on dilution during long-term cooling and small break LOCA, too, right? So the issue, the concern that came from the staff was if you have loop seal so deep, it's almost reached the mid-plane of the core at the bottom of the loop, slightly above.

And then you have this loop seal, then you

have a suppression of this situation. Then what's going to happen when you really discharge? That slug of fresh water from the loop seal to the core, that's a criticality situation.

I think on the address of issue, and this was not analyzed specifically for APR1400. And then we checked as a part of the regulatory check, because that's part of TMI action for each new design has to address that point, and we asked for version analysis.

And they performed the analysis and addressed this issue. They indicated there was no problem with one slug coming into the core, which caused the reactor to reach critical. There is no problem there. And the core will remain, you know, sub-critical during the long-term cooling of the small break LOCA.

And the basis came from the PKL standard test for typical PWR scenario. It's not for APR1400, but it's close enough to determining there is only slug coming in. Not simultaneous four slug injection. All right.

So that part is closed and then additional analysis of responses were considered acceptable by the staff.

MR. SCHULTZ: Shanlai, what were these

tests that were done?

MR. LU: The PKL test was done by, I think it was done over Europe. And then this was particularly tested just as part of support of the EPR, EPR development.

And then by the lead developer, this is not done by KHNP. But the PKL was a standard benchmark tested by OECD, too. So many countries were participating in the testing, both post-testing and pre-testing analysis, trying to figure out whether --

MR. SCHULTZ: To determine the dynamics of the slug release?

MR. LU: Exactly, exactly. Which is one part of the test program -- there is only one issue of that test that I was trying to address.

And that test did show that they have never observed simultaneous discharge after you have actuation of the natural separation, how come you have simultaneous? It's a chaotic situation.

Anyway, you have slight pressure difference. Now you're going to have all simultaneous release of the discharge of the slugs there.

So that's the basis that, I think, applied a reasonable, still conservative, assumption to assume what amount of water will be discharged into the core

and what kind of mixing and whether that would cause the reactor to cool. And their conclusion is no. The margin is maintained. And the staff agrees with that.

MR. SCHULTZ: Thank you.

MR. LU: Okay. Next item. Okay. This is part of GSI-181, and as I mentioned, that we are going to talk about more details there, but I'll give you a summary.

The LOCA deposition model, it's not an issue to really have a specific concern about the LOCA DM because we have a clean plant. It's important, 7.6 grams, something like that, per assembly fiber. It's not even an issue, a concern issue for the in-vessel downstream effects.

However, we still asked them to perform the LOCA DM analysis and that to show the crud formation for the given such a little amount of the break transported to the core what might be the problem.

And they provided the analysis already actually at the time we gave the presentation last time as a part of history. We are to finish this part of the review.

We just did not have the chance to put into the ACRS for you guys to review. So we declare

it still as an open item. But based on the review, they followed the -- they approved Westinghouse LOCA deposition model properly. And then we performed a confirmatory analysis based on the spreadsheet.

And we have what is the PWR Owner's Group. And their calculated temperature is less than 800 degree Fahrenheit for long-term cooling. So therefore, it's not an issue.

And so if you want to hear more, we can talk more in June if you really want to dive into more details. I think that's the time. But at this point, we don't see a problem there approving their RAIs's responses.

Okay. Next item. Loop seal clearing. I just mentioned about this one for the front dilution perspective, but from the PCT perspective. Right at the beginning of small break LOCA for only a narrow spectrum of the small break.

This reactor may get into the suppression of the two-phase water level of the core and because the loop seal is ready to be discharged and is pushed by the pressure in the steam generator.

And so there are two particular scenarios here. One is that very short, right at the beginning during the lowdown, right, or during the raffled or

there is another case of potentially there is a long term cooling there is a, you know, reformation of loop seal and re-discharge. So there are two points here.

So for their case, they analyze that one. That's part of the reason they are adding more spectrum. They were able to capture that narrow branch of the break size, which casts that initial loop seal clearing and then the core two-phase level suppression.

And they did the analysis. And they also analyzed, which was not done initially for the long-term cooling phase. And they performed additional small break LOCA analysis for a long time all the way to the annual established water level in a long-term cooling.

And then when the second loop seal formed and discharged, it's much, much less than 800, 600 or even 500 degree Fahrenheit. It's just a slow, very small break there.

So with that one and we also considered their RAIS responses adequate and issue a result.

Next slide. So in conclusion, here are the conclusions now. And all 10 open items were identified one year ago as part of presentation to ACRS subcommittee and were all resolved.

And the staff already approved and based on our SCR and already approved all their RAIS responses. They have done very good job to address the NRC staff concerns, the RAIs and then the issued have already been closed.

So from design perspective, Chapter 15 is done. We're done. Okay.

CHAIRMAN BALLINGER: Thank you. So this is going to be a great time for lunch. But before we do that, we need to see if there are any public comments.

So are there any members of the public in the audience that would like to make a comment? And I don't know if the line is open or not -- it's open? So are there any members of the public on the public line that would like to make a comment?

We always go through this exercise of trying to find out if anybody is out there that is actually listening. If you're out there, can you make a noise that says I'm here? Okay.

Having heard no noise -- you had something?

MR. SISK: This is Rob Sisk, Westinghouse. Just when you get a chance, I do want to respond back to the question of the swelling. I could probably do

it in the public session rather than the closed section so we can respond to that.

CHAIRMAN BALLINGER: Okay. That's fine. Do you want to do it now?

MR. SISK: Sure. We can do that. In consultation with KNF and KHNP just to clear up the question this morning, the question had to do with the swelling of SEA.

That operating experience that was being cited in the RAIS was Palo Verde having a swelling due to the B4C and the C.F. And the result of that operating experience was that they found that the swelling had a propensity after 10 years.

So what APR1400 does is they swap out their SEAS every 10 years to preclude potential. And operating history so far has shown that's been totally successful in resolving any concerns relative to SEA swelling to B4C.

CHAIRMAN BALLINGER: So they have experience with OPR 1000 then or some other experience in Korea with plants, with swapping these out every 10 years?

MR. SISK: No, that's their process. Let me put it this way. Yes, the answer to your question is yes.

CHAIRMAN BALLINGER: Okay.

MEMBER REMPE: Thank you.

MR. SISK: Thank you.

CHAIRMAN BALLINGER: Okay. So we should ask the members if there are questions that they have before we recess for lunch.

MEMBER SKILLMAN: No further questions, thank you.

CHAIRMAN BALLINGER: Anybody? Okay. We will recess for lunch. Come back at 1:30. The sessions we will be working on will be closed sessions then. So we will recess until 1:30.

(Whereupon, the matter went off the record at 12:29 p.m. and resumed at 3:37 p.m.)

CHAIRMAN BALLINGER: Okay. We're back in session. We remind you that it's an open session now and so we're going to try to get through the two pieces of your presentation on Chapter 3. If it takes beyond 5:00, we'll just stop and go on tomorrow morning.

So proceed.

MS. TERRY: Good afternoon, ACRS members. My name is Tomeka Terry. I am the Chapter PM for APR1400 design certification application review for Chapter 3, Design of Structures, Systems, Components,

and Equipment of the Advanced Safety Evaluation Report with no open items.

The staff provided a list of the technical staff who will present this afternoon to ACRS. In Section 3.1 through Section 3.61, 3.6.5 and Section 3.13 there were no specific issues in these sections. So, however, if ACRS members have any questions, the staff will be happy to answer your questions.

Now I'll turn it over to Renee Li will present Section 3.6.2.

MS. LI: Good afternoon. I'm Renee Li from Mechanical Engineering Branch.

The topics I'm going to discuss is determination of rupture locations and dynamic effects associated with postulated rupture of piping. The objective of the NRC staff review of this section is to verify and to ensure that adequate protection have been provided against the effects of the postulate pipe rupture and in compliance with the GDC 4 requirements.

The NRC staff reviewed the applicant's criteria used to define the pipe break/leakage crack location and the configurations including the break exclusion area. The staff also reviewed the applicant's methodology included in a technical

report; I would refer as TR in my later presentation, for addressing the dynamic jet impingement and blast wave effects. In addition, the NRC staff reviewed the pipe break hazards analysis summary report. I will refer as PRHA summary report.

With the resolution of the open items that I will discuss later the staff's overall conclusion on the review of this topic are: the applicant's criteria to define the pipe break/leakage crack location and the configurations including the break exclusion area are found acceptable.

Next slide. The TR methodology in addressing the dynamic jet impingement and blast wave effects are found acceptable. Also, the results of the PRHA summary report demonstrate that the APR1400 design is in compliance with the GDC 4 requirements and are acceptable.

Next I'll discuss the open item resolution. The first issue related to break exclusion area. the APR1400 DCD design provision for break exclusion area are consistent with the staff guidance. However, the break exclusion area are beyond the containment penetration area. Therefore, the applicant was request to justify the APR1400 DCD break exclusion area as described in the DCD.

In resolving this issue the applicant submit a RAI response. In that RAI response the applicant explained how the DCD break exclusion design provision were considered and applied to the results of the design of piping in the break exclusion area. In addition, the APR1400 design ensures sufficient accessibility to perform a 100 percent volumetric inservice examination of the pipe welds within the break exclusion area.

Next slide. Moreover, the results of the calculated maximum stress range were low compared to the relevant BTP 3-4 stress limit for postulating break locations. The RAI response also include markup of Revision 1 to applicable DCD Section 3.6.2 subsections.

Based on the review of the information provide by the applicant the NRC staff found that the applicant has adequately demonstrate its design provision and specifying a 100 percent volumetric inservice examination criteria meet the applicable BTP 3-4 break exclusion criteria in the NRC staff guideline. The applicant has appropriately justified the acceptability of expanding the break exclusion area beyond the containment penetration. In addition, the NRC staff found the proposed DCD markup include in

the RAI response were acceptable. This item is being tracked as a confirmatory item.

Next slide. The second open issue --

MEMBER SKILLMAN: Renee, would you back up two slides, please? The conclusion that you cite at the bullet at the bottom, the design ensures sufficient accessibility to perform a 100 percent volumetric inservice inspection of the pipe weld. At this early stage, the plant not having been built, without access to real isometrics, how could you make that conclusion?

MS. LI: Okay. I think it should be the design commitment in the DCD.

MEMBER SKILLMAN: Yes.

MS. LI: Right. Yes.

MEMBER SKILLMAN: And I think that that is captured on your next slide. In the first sentence you've identified the criteria for that. Thank you. All right. Thank you.

MEMBER REMPE: I have a question on this that was actually motivated by a comment from Member Vesna, but isn't this more than the inspections from this ASME standard, or like you were talking about earlier, oh, they're just doing what the ASME requires?

CHAIRMAN BALLINGER: I think Pete can probably -- I hope he's on the line, can probably say something, but --

MEMBER REMPE: Or maybe the staff knows, yes. I don't know if Pete's out there, but -- go ahead.

MR. REICHELT: Good afternoon. My name is Eric Reichelt. I think I can help out. I know that there was some questions earlier in the day about the inspectability for these new designs. No. 1, for pre-service inspection the COL is going to be inspecting in accordance with the requirements and guidelines of Section 3. Okay? And they've committed to performing 100 percent of the welds, which is a good thing both for them and for us because that provides some kind of a baseline further on down the line when they get into ISI space.

No. 2, when they perform the ISIs over the course of a 10-year interval when the plant is down they're going to be inspecting in accordance with the guidelines addressed in Section 11. Okay?

I think to answer your question about accessibility, the staff -- I attend -- I'm one of the members of the ASME Code, or one of the representatives from the NRC staff that attends the NS

-- the ASME Code meetings on a quarterly basis. And over the last 10 years we have been very vocal in our meetings about having -- making sure that -- and we're talking particularly with the utilities that, just like what you were saying, the plants are not built yet, so make sure that you have accessibility for 100 percent volumetric examinations. And in fact there's a new ASME Code Committee that addresses inspectability and accessibility. So I hope that helps.

MEMBER SKILLMAN: Thank you.

MS. LI: Okay. The second issue related to the PRHA summary report. To support the NRC staff's safety determination on the APR1400 PRHA assessment the applicant was request to submit summary information on the PRHA results to demonstrate that the APR1400 design meets the GDC 4 requirement.

In resolving this issue the applicant submit a RAI response with an associate PRHA summary report. In that report the applicant described the criteria used in determining postulate pipe failure location and the methodology for assessing the jet impingement and pipe whipping effects. Associate with those postulate high-energy line pipe breaks. The report also summarize the postulate break locations,

the target impacted by the respective break, as well as their associate protection method.

Based on the review of the information provided by the applicant the NRC staff found the PRHA report provides sufficient information to demonstrate that the PRHA were performed in accordance with the methodology and the criteria described in the APR1400 DCD and are in conformance with the applicable SRP for which the NRC staff found acceptable.

The results present in the report demonstrate that the APR1400 design is in compliance with the GDC 4 requirement such that SCCs important to safety are designed to accommodate and protect against the effects of postulate pipe failure. This item is closed.

Next slide. The third issue related to blast wave and potential feedback amplification and the resonance effects. The applicant was request to address the blast wave and potential feedback amplification and resonance effects as presented in a TR initially submitted for the staff's review. The applicant indicate that it would submit a TR revision with alternative approach to address these two issue.

Subsequently, the applicant submit a revised TR to address the blast wave effects. The TR

described that computational fluid dynamics, the CFD modeling, was performed for some of the most limiting steam break location. The CFD analysis approach was benchmark against several experiments and analysis of similar condition to verify its suitability. The results of the APR1400-specific CFD analysis were used to develop a methodology for assessing the blast wave effects for other HELB location that have not been analyze by CFD. The methodology conservatively accounted for the initial energy available to form the blast wave and the effects of plant geometry on the blast wave loading on the impact SSCs.

Next slide. To address the potential feedback amplification and resonance effects the applicant discuss in the TR how the multiple physical characteristic; for example, lack of perpendicular flat surface of sufficient size to establish a feedback loop, irregularity in contours of broke pipe end that distort the outgoing jet, of the APR1400 postulated HELB would prevent occurrence of a potential feedback amplification and resonance. The applicant has also show how the absence of HELB resonance effects was substantiate through a survey of experimental results.

Base on the review of the information

provide in the TR the NRC staff found that the applicant's CFC analysis perform for assessing the blast wave effects for the APR1400 plant are technically justified. The applicant has provided sufficient information to demonstrate the validity and the applicability of the test data and the methodology of the referenced open literature. The applicant has demonstrate that the methodology to account for effects of APR1400 plant geometry on the blast wave load is conservative. The applicant has adequately addressed the NRC staff's concern on the blast wave effects as identify in Appendix A of SRP 3.6.2. This issue was therefore close.

Next slide. In addition, the NRC staff found that the applicant has adequately discussed how the multiple physical characteristic of the APR1400 postulated high-energy line break would prevent occurrence of a potential feedback amplification and resonance. The applicant has also demonstrate that the absence of potential feedback amplification and resonance was substantiated by a survey of experimental results.

The NRC staff found the applicant's evaluation and approach to address potential feedback amplification and resonance effects as identify in

Appendix A of SRP 3.6.2 acceptable because the applicant has adequately demonstrate reasonable assurance that this phenomenon is not a concern for the APR1400 plant. This issue was closed.

This conclude my presentation.

CHAIRMAN BALLINGER: Questions?

(No audible response.)

CHAIRMAN BALLINGER: Next?

MR. REICHELT: Good afternoon. My name is Eric Reichelt. I am the reviewer for Section 3.6.3, Leak Before Break for the APR1400 DCD review.

I would like to also introduce Jay Wallace from the Office of Research who's in our -- in the audience who has provided us with technical assistance with the confirmatory analysis of the applicant's piping evaluation designs.

Next slide, please. For a leak before break the staff reviewed the applicable APR1400 DCD sections in 3.6.3., reviewed the DCD references for applicability and use. We held public meetings with KHNP and KEPCO staff about technical issues and RAIs leading to proposed DCD markups. The staff found these DCD sections acceptable. The technical issues and response to RAIs by KHNP were acceptable and were therefore closed. RAI 525-8685, question 3.6.3-9 was

an open item since the last time we met with the ACRS. That issue was the staff requested the applicant to review conflicting values between PICEP input and the DCD. And additionally we asked the applicant to provide copies of the PICEP input files for -- in order for us to complete your confirmatory analysis.

The applicant complied with our request. The staff reviewed the revised input values and calculations and received the PICEP Code from the applicant. The staff performed a confirmatory analysis of the applicant's piping evaluation designs for the applicant different lines that are addressed and that are identified in slide 13.

Next slide, please. Based on proprietary information provided by the applicant the staff was able to perform the confirmatory analysis of the leak before break piping evaluation diagrams. The NRC staff concludes that there is reasonable assurance the applicant's LBB analysis bounds the normal operation and normal operation plus safe shutdown earthquake design conditions and finds the applicant's LBB analysis acceptable. There are no open items for this section.

Next slide, please.

MR. SCHULTZ: Eric, this was -- excuse me,

this was a proprietary code that was provided to you by the applicant so you could do some studies?

MR. REICHELT: The PICEP Code is an EPRI-generated code, but it was used by the --

CHAIRMAN BALLINGER: I think it's actually written by Structural Integrity.

MR. SCHULTZ: That makes sense.

CHAIRMAN BALLINGER: Structural Integrity, but it's ran on a Commodore 64.

MR. REICHELT: But the values that were provided were provided in a proprietary table and --

MR. SCHULTZ: Did they provide the code or --

MR. REICHELT: Both.

MR. SCHULTZ: Okay.

MR. REICHELT: Both.

MR. SCHULTZ: And then what did you do in a confirmatory manner? What did you do with that information?

MR. REICHELT: Well, I'll let --

MR. WALLACE: My name is Jay Wallace. I'm from Office of Research. I provided the confirmatory analysis of their LBB. The PICEP Code that we received was the executable code, not the source code, so all we could do is run it with their values to see

what values we got. That came on the heels of a difficulty in reproducing their values. They had provided plots of crack leak versus COD for their desired leak rate by GPM. We couldn't reproduce those and that's what led to the RAIs initially. All of the rest of the values were proprietary values for the coefficients and so on that we didn't input into the code.

MR. SCHULTZ: So did you sensitivity studies or just validate what they had developed?

MR. WALLACE: We did not do any sensitivity studies. We ran the confirmatory analysis with our code --

MR. SCHULTZ: Okay.

MR. WALLACE: -- that is based upon ultimately XLPR Code. That's a completely different set. We didn't use PICEP or -- they had also used EPRI COD calculations. We have our own COD calculations as well as our own Leak Rate Codes. So it was absolutely independent of the work that they did, only using their values.

MR. SCHULTZ: Thanks for the additional explanation. Thank you.

MR. REICHELT: In conclusion the staff evaluation concludes on a design-specific and piping

system specific basis that the acceptance criteria are satisfied and therefore that dynamic effects of pipe rupture may be eliminated from design consideration.

This concludes my presentation. I will turn it over to Tom.

MR. SCARBROUGH: Okay. Good afternoon. I'm Tom Scarbrough. I'm going to summarize several sections that we reviewed in part of the DCD in Chapter --

(Simultaneous speaking.)

MEMBER SKILLMAN: Tom?

MR. SCARBROUGH: Yes?

MEMBER SKILLMAN: Excuse me --

MR. SCARBROUGH: Sure.

MEMBER SKILLMAN: -- for interrupting.

Eric, can we go back to your conclusion, please? The wording on your conclusion is so specific it tells me or communicates to me that only if the piping in the as-built is precisely identical to what is in the DCD is your conclusion accurate. Is that what you intend?

MR. REICHELT: In order for the COL -- since the -- let me see how I can phrase this for you. The -- for a COL, once he comes in with an application, okay, he is going to have to perform the

leak before break on his own piping layout. Obviously right now the piping layouts are not there. But the COL is going to have to perform an LBB with the applicant -- what -- in -- what the applicant did in the DCD. Okay? So the assumptions, the values, what they do need to be bounding to the applicant's DCD LBB analysis.

MEMBER SKILLMAN: Is that captured in a COL item?

MR. REICHELT: Yes, it is.

MEMBER SKILLMAN: Thank you.

Thank you. Excuse me, Tom.

MR. SCARBROUGH: So we're going to talk about several sections in 3.9. The first one is special topics.

And if you could go to the next slide, please. This covers several areas in regard to design transients. The staff found that the APR1400 transient occurrences are conservatively designed, based on the certified System 80+ transients and it was reviewed as part of that to verify that in the inservice conditions and such. There was one question about Level C, and what was determined was that was moved up basically to Level B to make it more conservative.

With regard to computer programs the staff conducted audits of a large number of computer codes and programs that were used for verification and validation, V&V. Had some follow-up questions. There was a follow-up item. There was one specific computer program that was identified that wasn't very well discussed, the DPVIB computer program, which is used to calculate fluctuating pressure distribution in a down-comer region. The staff went back and did a separate audit of that computer program and ensured that there was benchmark problems and evaluated those benchmarks and prepared a follow-up audit report. So with that the staff determined that the computer programs were acceptable and were adequately verified and validated.

With respect to faulted conditions, the staff reviewed the DCD and determined that the faulted conditions were in accordance with the Boiler and Pressure Vessel Code, Section III, Appendix F, which deals with service loadings for Level B service limits.

So the next section is 3.9.3, and this covers Code Class 1, 2 and 3 components, supports and core support structures. The staff concluded that the load combinations for ASME Code Class 1, 2 and 3

components and component supports conform to the ASME Boiler Code, Section III.

With respect to dynamic system loading such as pipe breaks and the POSRV actuation the staff found they conform with the Boiler Code as well as the dynamic fluid loadings such as valve actuation and water hammer conformed with Section III.

With respect to component supports the staff found that they will be designed in accordance with the Boiler Code, Section III, Subsection NF, which addresses the supports for nuclear reactors.

With respect to audits the staff has conducted two audits on design specifications to determine the design criteria, analytical methods and functional capability for the ASME Code and to confirm that the design information of DCD was translated into the design specifications.

Our first audit was in the August 2015 time frame. We found a number of areas that needed to be addressed in various components: reactor vessel valves, pumps, components, supports and restraints, and we provided a long list of issues that needed to be addressed in those specifications and also some areas actually where the DCD needed to be updated.

We conducted a follow-up audit and

prepared an audit report and found that for the most part those specs had been updated, however, there were some places where the specs that we were provided were not the formal signed versions and there still needs to be sufficient work on that.

So to be able to track that we issued an RAI to request that KHNP notify the staff when those specifications and DCD changes were available. And then specified a confirmatory action item for KHNP -- for us to look at those specs and ensure that those changes were made. So it's a confirmatory item now. I believe it was an open item when we visited you before, but it's a confirmatory item now, I believe we verified that. Okay? So that's 3.9.3.

3.9.6 is the Functional Design and Qualification and Inservice Testing Program. With respect to this section we evaluated Section 3.9.6 to make sure that using the Standard Review Plan, Section 3.9.6 -- we did ensure that the DCD specifies the application of ASME Standard QME-1-2007, which is endorsed in the NRC Reg Guide 1.100, Revision 3 for the functional design and qualification.

When we conducted the audit of the design specifications we ensured that the QME-1 standard was specified in those specifications. There were a few

places we had to have some comments back to sort of tighten up that discussion. For example, there are places where it might have must mentioned difference of pressure in terms of the testing for design qualification, and QME-1 specifies flow as well. So we wanted to make sure that it was clear that QME-1 would be followed. And so that was part of the audit and in the follow-up audit to make sure that was addressed.

We also noted that the IST Program has specified that OM Code with additional adjustments that we found from our review and had RAI responses, had a number of RAIs. And we reviewed -- since we talked to you last we went back and looked at DCD, Revision 1 and found that those RAI responses have been incorporated into DCD. And they cover things like the program description, making sure it was more clear, the testing of the power-operated valves, the motor-operated valves. And then there were a lot of changes that needed to be made in the IST table. We verified that those had been made to make sure it's consistent with the OM Code. So we've done that.

What's left regarding Section 3.9.6 is we have some planned ITAAC changes to reflect the latest version, and we have started going through Revision 2

of the DCD and found that, yes, that ITAAC is referencing QME-1, included the -- make sure that the proper qualification was made for those. That's an example of what we've gone back and checked on those ITAACs.

We also have -- as I mentioned, we have a confirmatory item that we're tracking for the design specification follow-up audit items and will be verifying that those changes were made and those specs were finalized when we are notified that they're available for us to review.

And then lastly we had the 3.9.6-1 question That had to do with a number of updates to the IC Program description and also the planned IST table. We verified those had been made in Revision 1 of the DCD and we were able to close that as well.

So that completes my presentation. If there are any questions -- if not, I'll pass along to Yuken Wong.

MR. WONG: My name is Yuken Wong from the Mechanical Engineering Branch. I'm the reviewer for Section 3.10, Seismic and Dynamic Qualification of Equipment. I will also present Section 3.9.2 and 3.9.5.

Section 3.9.2, Dynamic Testing and

Analysis of Structures, Systems and Components. The staff reviewed the methodology, testing procedure, inspection program, and dynamic analyses to ensure the structural integrity and functionality of piping systems, mechanical equipment, and their supports under vibratory loading.

Specifically, the staff reviewed the following six areas: Piping vibration, thermal expansion and dynamic effects testing during initial start testing for ASME Class 1, 2 and 3 piping; seismic analysis and qualification of seismic Category 1 components; dynamic analysis of reactor internals under steady-state and transient conditions; pre-operational flow-induced vibration testing of reactor internals; dynamic analysis of the reactor internals under faulted conditions; and lastly, correlation of reactor internals vibration test results analytical results.

Next slide. The staff reviewed the comprehensive vibration program, or CVAP report for the APR1400 steam generator and reactor internals design in comparison with the System 80 design. the Palo Verde Unit 1 reactor internal design is classified as the prototype and the APR1400 is the non-prototype Category 1.

One open item regarding the basis for using 32 hertz instead of the zero period acceleration of 50 hertz for APR1400 to determine if the equipment is rigid or flexible for selecting the static or dynamic analysis method. The other issue of this open item is regarding whether the fluid structure interaction and sloshing effects are included in the tank analysis. The applicant revised the DCD to use 50 hertz to determine if the equipment is flexible or rigid and confirmed that hydrodynamic forces exerted by the fluid on the tank walls will be included in the analysis. The staff finds the response acceptable and the open item is closed.

The other open item is regarding the benchmarking on the DPVIB Code used for the pump position pressure analysis as discussed earlier. The staff audited the benchmarking information for the computer code as part of the Section 3.9.1 review. The applicant provided a description of the code in Revision 1 of the DCD. The staff finds the response acceptable and the open item is closed.

Section 3.9.5. Next slide, please. Reactor Pressure Vessel Internals. The staff evaluate the arrangement of the reactor internals, their functions, flow paths through the reactor vessel, and

the design criteria. Compare the APR1400 reactor internal design and the CE System 80 reactor internal design; that is the Palo Verde Units 1, 2 and 3, because the reactor internals designs are similar.

The staff verified the core support structures and internal structures are designed and constructed in accordance with ASME, Section III, Subsection NG.

Next slide, please. One open item is regarding the structural integrity of control element guide tubes. KHNP provided calculations to show the guide tube structural integrity can be maintained during a safe shutdown earthquake. The applicant also reviewed the operating history of Korean plants with control element assembly and guide tube designs similar to the APR1400 designs and have confirmed that no failure have occurred on these components that would prevent control rod insertion. The staff finds the response acceptable and the open item is closed.

The other open item is about the seismic category of reactor internals. KHNP provided clarification that all reactor internals including internal structures are classified as Seismic Category 1. The staff finds the response acceptable and will confirm the change in the next revision of the DCD.

Next slide, please. Section 3.10, Seismic and Dynamic Qualification of Equipment.

The staff reviewed DCD Section 3.10, 3.7B.7.4, and technical reports related to equipment qualification; verified that equipment seismic qualification standards and methods are in accordance with Reg Guide 1.100, IEEE 344, and ASME QME-1; verified the procedures to evaluate effects of hard rock high-frequency response spectras on high-frequency-sensitive equipment. The staff conducted an audit of procurement and design specifications to verify seismic qualification methodologies are in accordance with the DCD.

There is a confirmatory item regarding the audit findings. In the follow-up audit the staff verified that the applicant's resolution of the staff's findings on the first audit are acceptable. The staff will verify the updated procurement specifications to address the staff findings.

This concludes my presentation.

MS. LI: Okay. So we have the next three people to go next, and I think we probably will finish before 5:00.

CHAIRMAN BALLINGER: Good, good.

MR. CINTRON-RIVERA: Good afternoon. My

name is Jorge Cintron. I'm an electrical engineer from the Electrical Engineering Branch in NRR, and I'm the lead reviewer for Section 3.11, Environmental Qualifications of Electrical and Mechanical Equipment.

Section 3.11 provides the APR1400 approach for environmental qualification of mechanical and electrical equipment. The staff reviewed the environmental qualification of mechanical and electrical equipment to verify that the equipment is capable of performing during functions under all environmental conditions, anticipated operational occurrences, accident and post-accident environmental conditions. Some of the equipment that is required to be qualify is the safety-related equipment, non-safety-related equipment, supply and safety-related require, and certain post-accident monitoring condition.

For the purpose of this presentation we'll be discussing an open item that we have from the previous presentation.

The staff evaluated the applicant's response to RAIs requesting justification of the use of IEEE 323-2003. Regulatory Guide 1.89 provides an acceptable criteria for qualifying electrical equipment and endorsed the IEEE 323-1974. The APR1400

deviates from the Reg Guide and is proposing to use the 2003 version.

The staff perform a comparison on both standards and has requested the applicant to provide a justification of why the 2003 version is acceptable and how to tell the technical differences between both them.

Next slide. The staff evaluated the applicant's response and concludes that the applicant has appropriately justified the technical differences between the IEEE 323-2003 and IEEE 323-1974. The staff has completed the applicant's -- completed the evaluation, and therefore this is a confirmatory item.

That concludes my -- the part of my presentation.

MR. STUTZCAGE: All right. Hi, I'm Ed Stutzcage. I'm the reviewer for the radiological section of Section 3.11.

Next slide, please. These were the open items we had last time. There were several open items associated with these two questions. These are the issues that we discuss in our next slides.

Next slide, please. The first one was inconsistencies with Chapter 12 information regarding the neutron dose in the containment building operating

floor during normal operation. There was additional information needed to ensure that the total integrated neutron dose on the operating floor was adequate. And I think we had a -- partially from an ACRS question last time we asked them if they could provide some operational data for the dose in the operating floor area. And they did. They provided operational data from the Shin Kori Unit 3 plant and it was significantly lower than the dose calculated in the EQ analysis. So as a result of that the staff found it to be acceptable.

Next slide, please.

MR. SCHULTZ: Everything is going to be a lower dose.

MR. STUTZCAGE: Yes, that's true. That resolved that item.

The next one was the doses within the auxiliary building during accidents did not appear to adequately consider radiation streaming through the containment penetrations.

The doses in areas with containment penetrations were calculated at dose point at the center of the room. So this is for rooms that had containment penetrations in them. The doses were calculated in the center of the room. We asked they

justify why that was acceptable because not all the equipment is going to be in the center of the room. And as a result they provided -- they revised the calculations for those rooms provided the calculated TIDs with the -- at a 0.30 centimeters away from the penetrations. Staff found that to be acceptable and that resolved this issue.

Next slide, please. There was a couple other areas where the total integrated dose calculations weren't making a lot of sense to us, so we asked a few additional questions and they provided some additional information. For example, rooms that didn't have radiation sources within them, they had -- they were affected by sources in different rooms. They provided some additional explanation. They indicated that they updated the technical report describing that the total integrated dose for areas without radiation sources were calculated described -- calculated assuming one percent failed fuel defects from adjacent sources with additional margin added for conservatism. It's consistent with the guidance and is acceptable.

The applicant also revised the calculations in the transfer tube inspection area to account for fuel transfer, which was -- the dose went

from being almost very low to extremely high accounting for fuel transfer. And they also provided some additional updates clarifying shielding thicknesses for certain areas near and around demineralizers and filter areas.

Next slide, please.

MR. SCHULTZ: If you consider all of these areas that were addressed here, have they covered everything that you were interested in?

MR. STUTZCAGE: I believe so. I believe that, yes, they updated it to -- their calculations are now conservative or realistic for what you'd expect using the guidance.

MR. SCHULTZ: Good. So it wasn't clear from the bullets whether they got everything that you were --

MR. STUTZCAGE: Yes, they had addressed all the areas I had concerns.

MR. SCHULTZ: Good. Thank you.

MR. STUTZCAGE: Yes. The conclusion for this issue, the staff reviewed the shielding and dose rate information using microshield as necessary and found them to be acceptable.

Next slide. The last one was the assumption for post-accident fluid rate outside

containment. The applicant did not provide enough information explaining the allowed ESF leakage rate assumed in the EQ analysis during and accident and why the leakage was different than that used in the accident analysis. There's a difference between the Chapter 15 analysis and the Chapter 3 for the leak rate.

The applicant clarified that the ESF leakage rate for EQ was based on the assumed leakage rate for valves and pump specified in their response. The allowed leakage rate was doubled in accordance with Reg Guide 1.183. The applicant also proposed to update Chapter 15 and the technical report for EQ to clarify that what the dose rate was the basis -- that the leak rate was and the basis for it, and also what the basis for the Chapter 15 analysis was.

Next slide, please.

MR. SCHULTZ: You've got the number for the Chapter 15, but could you remind us of the basis for what they assumed in Chapter 15?

MR. STUTZCAGE: I'm not the Chapter 15 reviewer. All that I can say is it's based on a different amount of -- a more conservative leak rate than is used in the Chapter 3 analysis. The Chapter 15 analysis uses a higher number. That's all that I

could say right now.

MR. SCHULTZ: Any clarification? You may not have the right people here. I remember the comparison and I just -- it's more of a curiosity question. Looks like you've got the right relationship between Chapter 15 and this evaluation.

MR. STUTZCAGE: They assume --

MR. SCHULTZ: Maybe that's sufficient.

MR. STUTZCAGE: They assume different leak rates from different size valves and pumps and stuff, and it just -- they used different assumptions for Chapter 15 and Chapter 3, which is not typically their -- they use the -- most applications use the same assumptions. In KHNP's case they wanted to use different assumptions. We found them both to be conservative leak rates. And as I say in the next slide if during actual operation the leak rates are higher than what they assume here, they'd have to make sure they take appropriate action and reanalyze the qualification of the equipment or the -- or fix the leak to make sure they don't have -- it's such a large effect on the equipment.

MR. SCHULTZ: I think that's a good answer. I appreciate it. Thank you.

MR. STUTZCAGE: Thanks. That concludes my

presentation.

MEMBER POWERS: I wonder if I could just a question to the last two speakers, Jorge and Edward.

Not concerning this certification at all, but just in general, are we getting information coming from the damaged plants at Fukushima that would move us to reconsider our requirements for post-accident environmental qualification of safety-related equipment?

MR. STUTZCAGE: From my perspective I don't think that we're using any of the information from Fukushima for the radiological piece. I think we're still basing it on our Reg Guide 1.183 and Reg Guide 1.89.

MEMBER POWERS: Yes.

MR. STUTZCAGE: That's -- yes, I don't -- we -- we haven't changed anything I could think of --

MEMBER POWERS: Well, it's not changed. It is are we getting information that might move us to change?

MR. STUTZCAGE: I -- yes, I don't know the answer to that.

MEMBER REMPE: I feel obligated to speak up. I think industry is exploring that and seeing if they can develop a case, from interactions I have

doing other activities.

MEMBER POWERS: One of the things that we have to admit about our Environmental Qualification Program is that we distinguish and separate the radiological impact on the equipment and the thermal humidity impact. We've always known we've done that, but in our research that is the basis for our License Renewal Program we find that there are synergistic effects between radiation and -- so the question comes up is are we adequately conservative by making this distinction?

MR. CINTRON-RIVERA: Yes, as part of the open item that we have addressing the difference between both standards, that was one of the questions that we asked, because as you mentioned, that is required for synergistic effects, but the actual testing that we perform has been more separately. So we -- that was one of the questions that we asked and basically what the response that we get I -- we believe this provide us reasonable assurances that the qualification we performed, in a way that -- the sequence of the testing will be performed, that -- the one that provides the most degradation. So, and that was basically what industry is performing right now.

And as far as looking to the future, I

don't have much information of --

MEMBER POWERS: Yes.

MR. CINTRON-RIVERA: -- they'll now performing synergistic testings or --

MEMBER POWERS: One of the things that has arisen, and certainly following the accident at TMI, but maybe it's more evident now as we continue to look at Fukushima, is the potential for generating acidic gases in the environment, both hydrochloric acid and nitric acid. And that really is not recognized in our testing program except insofar as that which is being tested; for an instance, in the radiation environment, can generate some acid, but it doesn't have the benefit of the elevated temperatures and the humidity.

So it's of interest to me to know if from Fukushima we derive anything that would move us to change the way we look at environmental qualification. It has nothing to do with this particular application, but I'm just looking to the future to see if you guys had any insight on this. Sounds like you've at least thought about it, though.

Maybe I will just mention one thing our testing for radiological effects or radiation effects does not do is technically we test by just exposing them a gamma field. We never look at the effect of,

for an instance, radioactive aerosol depositing locally on wiring and things like that. Probably not a crucial thing for light water reactors, but as we move in these more exotic reactors: gas reactors, sodium reactors and salt reactors, then the deposition in the local hot spot on equipment might be a more important thing to consider.

MR. CINTRON-RIVERA: Okay.

MR. TSIRIGOTIS: My name is Alexander Tsirigotis and I'm a mechanical engineer for the Mechanical Engineering Branch in the New Reactors, and I'm the reviewer for the APR1400 DCD Section 3.12, which is associated with the ASME Class 1, 2 and 3 piping and its associated supports.

The NRC staff reviewed the structural integrity of the APR1400 piping and pipe supports and it was confirmed that the design of the piping and pipe supports incorporates NRC guidance and is in accordance with ASME Boiler and Pressure Vessel Code, Subsections NB, NC and ND for piping and NF for pipe supports, as incorporated by reference in 10 CFR 50.55a.

The staff conducted audits of pipe stress analysis and pipe support designs to confirm consistency with the DCD and that the structural

analysis meet ASME Section 3 acceptance criteria.

During the review of DCD Section 3.12 the staff held public meetings with the applicant to discuss technical issues that came out from audits and during the resolution of RAIs and which led to proposed DCD markups, revisions to technical reports and other piping analysis. The applicant has adequately addressed all of our -- all of the NRC staff comments.

During Phase 4 of the review the applicant addressed the structural effects on piping and its supports that could potentially occur from the operation of the safety injection tank and its fluidic device. The applicant derived water hammer loads which when incorporated in the load combinations for pipe stress analysis and pipe support design showed that piping and supports remained structurally adequate and met ASME Section III acceptance criteria.

To address the vibration on the safety injection line the applicant modified the safety injection tank subsystem test, DCD Tier 2 Section 14.2.12.1.22, to instrument the safety injection line to gather vibrational data and to ensure that vibration levels are within acceptable limits as found in ASME OM-SG Part 3. Part 3 of the Standards Guides

of the OM is for pre-operational and initial startup vibration testing of piping systems. And basically if the vibrational stresses are below the endurance limit of the material, they're found to be acceptable. And the applicant saw that.

Any questions?

(No audible response.)

MS. TERRY: That concludes Chapter 3's presentation for today. Any questions from the staff?

(No audible response.)

MS. TERRY: ACRS members?

(No audible response.)

CHAIRMAN BALLINGER: Thank you. I think this is a -- after we get comments from the public and go around the table this is going to be where we'll take a break until tomorrow morning.

So are there any comments from members of the public in the audience?

(No audible response.)

CHAIRMAN BALLINGER: There is no audience.

Pete?

MEMBER RICCARDELLA: Yes, I'm on.

CHAIRMAN BALLINGER: Right. Have you got any questions or comments?

MEMBER RICCARDELLA: No, no comments.

CHAIRMAN BALLINGER: Okay. Is there anybody else, anybody out on the public line that would like to make a comment?

(No audible response.)

CHAIRMAN BALLINGER: It always takes a few seconds for people to un-mute.

(No audible response.)

CHAIRMAN BALLINGER: Hearing none, the next thing to do is to go around the table and ask if there are comments from the members. I started over there earlier, so, Mike?

MEMBER CORRADINI: I don't have any comments. I'd just thank the staff and KHNP for their presentations. We through a lot of stuff today.

CHAIRMAN BALLINGER: Dick?

MEMBER SKILLMAN: I want to thank the staff and the KHNP for hearing comments that we made at previous meetings and weaving those comments into today's presentations, specifically the one that Alex just identified, the SIT vibration that I was so concerned about. But there have been a number of those through today's presentation that clearly identify that the staff and KHNP have heard some of the concerns that we've raised. So great presentation and thank you.

CHAIRMAN BALLINGER: Dana?

MEMBER POWERS: Well, I'll just reiterate one of the things that we need to look for as we consider the certification of more exotic reactors than conventional LWRs is this issue of the environmental qualification of safety-related equipment in the post-accident environment, because I think it will be different. And there are approximations, typically very conservative approximations that we've made in our environmental qualification requirements whose conservatism may come to question as we move into these more exotic reactors. I think it's an area that the ACRS may want to look at as part of their general considerations for the future.

MEMBER SUNSERI: I just extend my appreciation to the staff and KHNP for their presentations today. No other comments.

MEMBER STETKAR: Nothing more. Thank you.

MEMBER MARCH-LEUBA: I also don't have any comments.

CHAIRMAN BALLINGER: Charles?

MEMBER BROWN: No.

CHAIRMAN BALLINGER: Joy?

MEMBER REMPE: I just wanted to add my

thanks. I appreciate as always KHNP going and finding an answer if they don't have it right there for us.

MEMBER DIMITRIJEVIC: No comments. Thank you.

CHAIRMAN BALLINGER: Well, I'd like to -- well, Steve?

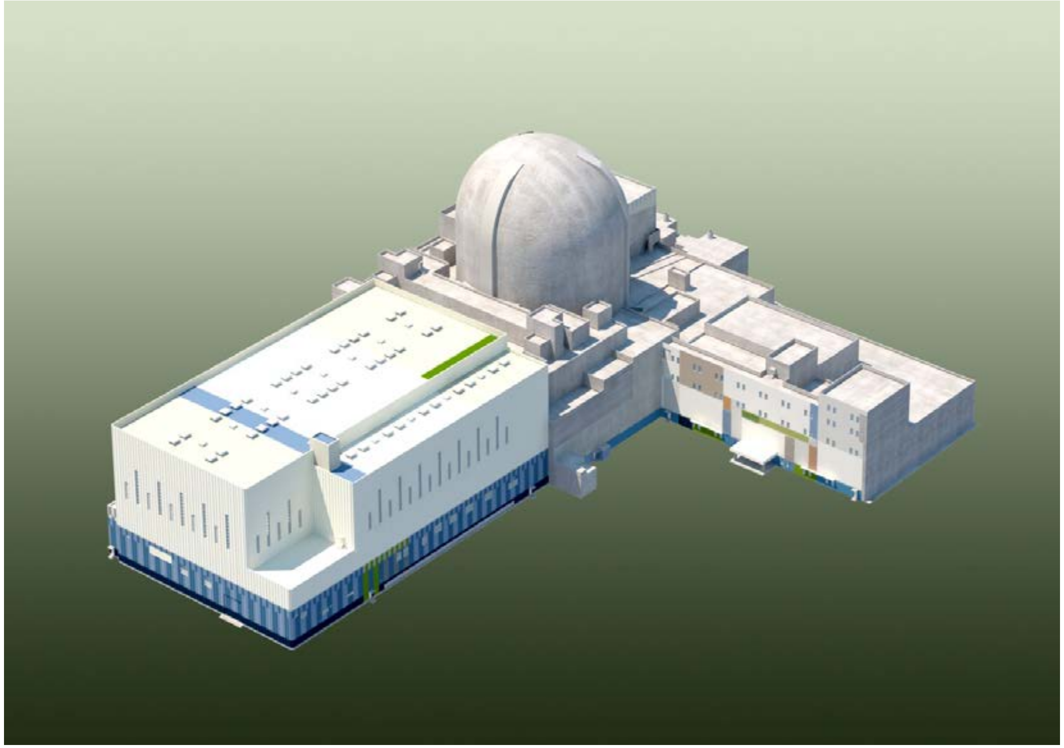
MR. SCHULTZ: I really do appreciate the presentations by both KHNP and the staff. The presentations really do represent substantial progress in the overall review by the staff and the presentations by the applicant. And I know the Committee recognizes the level of effort and depth of analysis that has been done to support the summary presentations that we've heard today, and the clarifications and answers to the questions today have been very helpful. Thank you.

CHAIRMAN BALLINGER: And I'd like to echo the comments of the other members and Steve. It was a long day and a lot of -- I can see the comments that were incorporated, as Dick mentioned. I counted like six, and I was keeping track. So we appreciate that a great deal. So that being said, we are adjourned until 8:30 in the morning.

(Whereupon, the above-entitled matter went off the record at 4:41 p.m.)

NON-PROPRIETARY

APR1400 DCA
Chapter 3: Design of Structures, Systems, Components, and Equipment



KEPCO/KHNP

April 17, 2018

Contents

- **Overview of Chapter 3**
  - Section Overview
  - List of Submitted Documents and Summary of RAIs
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- **Summary of Open Items**
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- **Attachments:**
  - Acronyms
  - List of COL Items related to Open Items

Overview of Chapter 3

Section	Title	Major Contents
3.1	Conformance with Nuclear Regulatory Commission General Design Criteria	• Conformance with Nuclear Regulatory Commission General Design Criteria 1 through 64
3.2	Classification of Structures, Systems, and Components	• Classification of Structures, Systems, and Components according to nuclear safety classification, quality groups and seismic category
3.3	Wind and Tornado Loadings	• Design features of wind and tornado/hurricane loading considered in the design of seismic Category I and II structures
3.4	Water Level (Flood) Design	• Design features of flood protection from internal and external sources considered in the design of seismic Category I and II structures
3.5	Missile Protection	• Design features of internally generated missiles • Design features of external missiles considered in the design of seismic Category I and II structures
3.6	Protection against Dynamic Effects Associated with the Postulated Rupture of Piping	• Design Protection against postulated piping failures in fluid system • Determination of break locations and dynamic effects associated with postulate rupture of piping • Design features of pipe whip restraints • Leak before-break evaluation procedure

Overview of Chapter 3

Section	Title	Major Contents
3.7	Seismic Design	• Seismic input motions • Seismic analysis methodology and results of seismic Category I Structures with generic soil profiles • Seismic analysis methodology of seismic Category I subsystems • Seismic monitoring system
3.8	Design of Category I Structures	• Design features of Category I Structures including concrete containment, internal structures of containment, other seismic Category I structures, and foundations
3.9	Mechanical Systems and Components	• Design, dynamic testing and analysis for ASME Code Section III, Division 1, Class 1, 2, and 3 components and supports including core support structures.
3.10	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	• Acceptance criteria, code and standards, procedures, and methods applied to the seismic and dynamic qualification of mechanical and electrical equipment including instrumentation
3.11	Environmental Qualification of Mechanical and Electrical Equipment	• Equipment Location and Environmental Conditions, Qualification Tests and Analysis, Environmental Qualification Method. • Equipment Qualification List, Environmental Parameters Data.
3.12	Piping Design Review	• Design of the piping system and piping support including the structural integrity, as well as the functional capability. • The design transients and resulting loads and load combinations with appropriate specified design and service limits.
3.13	Threaded Fasteners (ASME Section III Class 1, 2, and 3)	• Design feature of ASME Section III Class 1, 2 and 3 component fastener

Overview of Chapter 3

❖ List of Submitted Documents for Chapter 3 (1/2)

Document No.	Title	Revision	Type
APR1400-K-X-FS-14002 -P & NP	APR1400 Design Control Document Tier 2: Chapter 3 Design of Structures, Systems, Components and Equipment	2	DCD
APR1400-K-X-IT-14001 -P & NP	APR1400 Design Control Document Tier 1	2	DCD
APR1400-E-S-NR-14001-P & NP	Seismic Design Bases	2	TER
APR1400-E-S-NR-14002-P & NP	Finite Element Seismic Models for SSI Analyses of the NI Buildings	2	TER
APR1400-E-S-NR-14003-P & NP	SSI Analysis Methodology and Results of NI Buildings	2	TER
APR1400-E-S-NR-14004-P & NP	Evaluation of Effects of HRHF Response Spectra on SSCs	3	TER
APR1400-E-S-NR-14005-P & NP	Evaluation of Structure-Soil-Structure Interaction(SSSI) Effects	2	TER
APR1400-E-S-NR-14006-P & NP	Stability Check for NI Common Basemat	4	TER

Overview of Chapter 3

❖ List of Submitted Documents for Chapter 3 (2/2)

Document No.	Title	Revision	Type
APR1400-Z-M-NR-14009-P&NP	Comprehensive Vibration Assessment Program for the Reactor Vessel Internals	1	TER
APR1400-E-X-NR-14001-P&NP	Equipment Qualification Program	0	TER
APR1400-E-N-NR-17001 -P & NP	APR1400 High Energy Line Break Jet Impingement	1	TER
APR1400-E-N-NR-14004-P&NP	Summary Report of High-Energy Piping Rupture Analysis	2	TER
APR1400-Z-M-NR-14016-P&NP	Leak-Before-Break Evaluation of the Surge Line	2	TER

❖ Summary of RAIs

No. of Questions	No. of Responses	No. of OI
260	260	23

NON-PROPRIETARY

Overview of Chapter 3

❖ List of Open Items (1/3)

No.	Related RAI	Topic	ADAMS Accession #
1	RAI 41-7957 (Q 03.06.02-2)	Pipe rupture hazard analysis	ML17296A493
2	RAI 166-8198 (Q 03.06.02-3)	Break exclusion area	ML17271A449
3	RAI 359-8448 (Q 03.06.02-6,7,8)	Blast wave and potential feedback amplification and resonance effects	ML17237A145
4	RAI 525-8685 (Q 03.06.03-9)	Leak-Before-Break Analysis	ML17333A371
5	RAI 183-8197 (Q 03.07.02-1)	Hard Rock High Frequency(HRHF)	ML17242A320
6	RAI 208-8245 (Q 03.08.03-5)	Analysis Methods and Results for RCB Internal Structure	ML17233A379
7	RAI 248-8295 (Q 03.08.05-1)	Analysis and Design for Critical Sections	ML17157B583
8	RAI 255-8285 (Q 03.08.05-4)	Waterproofing Membrane	ML17026A465
9	RAI 255-8285 (Q 03.08.05-7)	Construction Sequence	ML17362A154
10	RAI 255-8285 (Q 03.08.05-8)	Design and Analysis Procedures for NI Common Basemat	ML17255A944

NON-PROPRIETARY

Overview of Chapter 3

❖ List of Open Items (2/3)

No.	Related RAI	Topic	ADAMS Accession #
11	RAI 255-8285 (Q 03.08.05-12)	Applied loads of NI Common Basemat	ML17251A159
12	RAI 255-8285 (Q 03.08.05-13)	Loads and Load Combinations for NI Common Basemat	ML17251A121
13	RAI 255-8285 (Q 03.08.05-16)	Subgrade Modulus of Soil Profiles for APR1400	ML17256A182
14	RAI 255-8285 (Q 03.08.05-17)	Differential Settlement of NI Common Basemat	ML17241A144
15	RAI 151-8078 (Q 03.09.02-9)	Pump-Induced Loads on RVI Components	ML16144A654
16	RAI 92-8068 (Q 03.09.05-6)	Control Element Guide Tube buckling integrity and operating experience	ML17255A934
17	RAI 92-8068 (Q 03.09.05-9)	In-Core Instrumentation Support System	ML16354A541
18	RAI 92-8068 (Q 03.09.05-16)	Loading Condition – Level D Service Loading	ML16306A443

Overview of Chapter 3

❖ List of Open Items (3/3)

No.	Related RAI	Topic	ADAMS Accession #
19	RAI 92-8068 (Q 03.09.05-18)	Bases for Reactor Internals - Classification	ML16354A541
20	RAI 69-7994 (Q 03.09.06-9)	Pump and Valve In-service Testing Program Table	ML16242A438
21	RAI 27-7944 (Open Item 03.11-1)	Environmental Qualification of Mechanical and Electrical Equipment	ML15198A260
22	RAI 527-8686 (Open Item 03.11-2)	Environmental Qualification of Mechanical and Electrical Equipment	ML17004A034
23	RAI 549-8856 (Q 3.12-19)	Structural integrity of piping and pipe supports that could be impacted by vibration and water hammer from the operation of SIT and FD	ML17332A108

Summary of Open Items

❖ Open Item: Pipe Rupture Hazard Analysis

- Related RAIs : 41-7957 (Q 03.06.02-2)
- Description of issue
  - Staff requested:
    - ✓ Assessment of the dynamic effects of jet impingement and blast waves and fatigue considerations in determining the postulated break locations in ASME Class 1 high energy piping systems.
    - ✓ Technical report for Pipe Rupture Hazard Analysis (PRHA) summary (APR1400-E-N-NR-14004-P&NP) incorporated jet impingement, blast wave, and resonance effects.
- Resolution:
  - ✓ KHNP provided the PRHA summary report incorporated the postulated break locations and the methodology for assessment of jet impingement and blast waves in October, 2017.
  - ✓ KHNP provided the text description regarding non-conservatisms of ANSI/ANS 58.2-1988, in revised response to RAI 41-7957 Q.03.06.02-2 in October, 2017.

Summary of Open Items

❖ Open Item: Break Exclusion Area

- Related RAIs : 166-8198 (Q 03.06.02-3)
- Description of issue
  - Staff requested to provide:
    - ✓ Acceptability of expanding the break exclusion area to the auxiliary building anchor wall beyond the isolation valve and justification with the results of pipe rupture analysis.
- Resolution
  - KHNP provided confirmation that calculated maximum stress results for the graded approach piping in break exclusion area were low compared to the SRP BTP 3-4 stress limit for postulated break locations in revised response to RAI 166-8198 in September 2017.

Summary of Open Items

❖ Open Item: Blast Wave and Potential Feedback Amplification and Resonance Effects

- Related RAIs : 359-8448(Q 03.06.02-6,7,8)
- Description of issue
  - Staff requested:
    - ✓ Clarification on the V&V of the CFD model for blast wave effects
    - ✓ Methodology for evaluation of blast wave
    - ✓ Justification of the criteria used for considering the effect of potential oscillatory jet loads

Summary of Open Items

- Related RAIs : 359-8448(Q 03.06.02-6,7,8)
- Resolution:
  - KHNP provided the technical report 'APR1400 High Energy Line Break Jet Impingement (APR1400-E-N-NR-17001-P &NP)' containing CFD analysis, V&V results and source of the literature references in June 2017.
  - KHNP provided that the feedback amplification and resonance effects due to high energy line breaks does not occur in APR1400 with thermodynamics and lack of perpendicular flat surfaces.
  - The NRC staff requested additional information or clarification for evaluating the dynamic jet impingement and blast wave effects to complete the staff's review.
  - KHNP provided the revised technical report included the requested clarification such as literature summary and CFD V&V in October 2017

Summary of Open Items

❖ Open Item: Leak-Before-Break Analysis

- Related RAIs : 525-8685 (Q 03.06.03-9)
- Description of issue
  - The PICEP input used for calculating the leakage crack length did not agree with the values stated in the APR1400 DCD.
  - KHNP was requested to confirm the input values and calculations in the DCD are correct and provide copies of the PICEP input files.
- Resolution:
  - KHNP revised the fluid temperature in the PICEP input and calculation.
  - KHNP provided the response with revised input values and calculation, and incorporated the results into related DCD and TeR (APR1400-Z-M-NR-14016-P & NP, Leak-Before-Break Evaluation of the Surge Line).

Summary of Open Items

❖ Open Item: Hard Rock High Frequency (HRHF)

- Related RAIs : 183-8197 (Q 03.07.02-1)
- Description of issue
  - Staff requested to provide an appropriate number of modes to capture and justification for ISRS reduction levels in excess of those provided in SRP Section 3.7.2 II.4.
  - Staff requested to provide revised results for ISRS comparisons and seismic loads between the CSDRS and the HRHF, and justification for the acceptability of all HRHF exceedances.
- Resolution
  - 16 modes were selected to capture converged incoherent motion. It was justified that this converged solution gave the ISRS reduction levels still exceeding the reduction limit set forth in SRP 3.7.2 II.2, as described in Technical Report (APR1400-E-S-NR-14004, Evaluation of Effects of HRHF Response Spectra on SSCs, App. C).
  - From the ISRS comparisons and re-evaluation of the seismic loads, the current designs of the RCB CS, RCB IS, and AB were found to have adequate margin, and thereby all HRHF exceedances were acceptable. The revised results were described in above Technical Report App. C.

Summary of Open Items

❖ Open Item: Analysis Methods and Results for RCB Internal Structure

- Related RAIs : 208-8245 (Q 03.08.03-5)
- Description of issue

  Staff requested to explain the analysis method under seismic loadings for the floor slabs between the secondary shield wall and the containment shell, and the connections between the floor slab and containment shell.

- Resolution
  - Response spectrum analyses were performed using the FRS which envelope both containment shell and secondary shield wall for the seismic load. A detailed procedure for design and analysis of the floor slabs were described in DCD Subsection 3.8.3.4.1, and the design results were provided in DCD Table 3.8A-43, Table 3.8A-44, and Figure 3.8A-61.
  - KHNP provided the connection details between floor slabs and containment shell in DCD Tier 2, Subsection 3.8A.1.4.3.4.3 and Figure 3.8A-60, those could allow the movement of both horizontal directions (tangential and radial).

Summary of Open Items

❖ Open Item: Analysis and Design for Critical Sections

- Related RAIs : 248-8295 (Q 03.08.05-1)
- Description of issue
  - Staff requested to add other design sections and to include their missing information.
  - Staff requested to provide the lacking design results of concrete structures with consistent format.
- Resolution
  - KHNP provided the information of selection criteria for design sections in DCD Tier 2, Section 3.8A. KHNP also added the missing design sections (dome, liner, slab, steel beam) and their design results in DCD Tier 2, Subsection 3.8A.1.4.
  - KHNP provided the information of design results for concrete structures where the information was lacking. Accordingly, Subsection 3.8A.1.4.3.2.3 and Table 3.8A-42 for IRWST structure, Subsection 3.8A.1.4.2.3.1 and Table 3.8A-10 for basemat structure were updated.

Summary of Open Items

❖ Open Item: Waterproofing Membrane

- Related RAIs : 255-8285 (Q 03.08.05-4)
- Description of issue
  - Staff requested to provide the description whether waterproofing membranes were used in APR1400 design.
  - Staff requested to provide the design effects of the waterproofing membranes, if used, on shear resistance of NI common basemat.
- Resolution
  - KHNP confirmed the use of waterproofing membranes and provided the information of typical details for installation, as shown in DCD Tier 2, Subsection 3.8.5.1 and Figure 3.8-27.
  - KHNP provided the sliding resistance effect associated with waterproofing membranes beneath basemat in the RAI response and DCD Subsection 3.8.5.1, in which the design parameter of coefficient of friction will be verified by COL 3.8(11).

Summary of Open Items

❖ Open Item: Construction Sequence

- Related RAIs : 255-8285 (Q 03.08.05-7)
- Description of issue
  - Staff requested to describe how the construction sequence and differential settlement of foundations were considered in the load and load combination.
  - Staff requested to provide the approach for four types of settlements (maximum vertical settlement, tilt settlement, differential settlement between structures, and angular distortion) based on analysis results.
- Resolution
  - The member forces and moments from the analyses with/without construction sequence were considered in dead load case of the design load combinations. The detailed descriptions were added in DCD Subsection 3.8.5.4.2.1.
  - The approach for assessment of four types of settlements and their results were provided in the DCD Subsection 3.8.5.4.2.2 and Technical Report (APR1400-E-S-NR-14006-NP, Stability Check for NI Common Basemat, Section 5). These four types of settlements will be verified based on site-specific soil conditions by COL 3.8(18) and COL 3.8(19).

Summary of Open Items

❖ Open Item: Design and Analysis Procedures for NI Common Basemat

- Related RAIs : 255-8285 (Q 03.08.05-8) :
- Description of issue
  - Staff requested to justify the use of equivalent static accelerations with 100-40-40 method considering phasing motion of three superstructures.
  - Staff requested to explain how to consider torsional load in the basemat analysis.
- Resolution
  - To consider phasing motion of three superstructures (RCB CS, RCB IS, AB), the results from both the linear SRSS combination on the member force level and the nonlinear method using 100-40-40 combination on the input level were enveloped. Additional detailed descriptions were added in DCD Subsection 3.8A.1.4.2.3 and Technical Report (APR1400-E-S-NR-14006-NP, Stability Check for NI Common Basemat, Section 3.2.5.)
  - The member forces and moments from torsional analysis of superstructures were added to the seismic load cases by absolute summation. Additional detailed descriptions are added in the above mentioned DCD Subsections and Technical Report .

Summary of Open Items

❖ Open Item: Applied Loads for NI Common Basemat

- Related RAIs : 255-8285 (Q 03.08.05-12)
- Description of issue
  - Under the KHNP response to RAI 255-8285 Q 03.08.05-8, Staff requested to provide the maximum bearing pressure resulting from both the linear SRSS combination and the nonlinear 100-40-40 combination to resolve phasing of the responses from each of the three superstructures when analyzing NI common basemat.
- Resolution
  - The bearing pressures for NI Common Basemat were re-calculated from both the linear SRSS combination and the nonlinear 100-40-40 combination. Enveloped bearing pressures were provided in the DCD Subsection 3.8A.1.4.2.3.5.

Summary of Open Items

❖ Open Item: Loads and Load Combinations for NI Common Basemat

- Related RAIs : 255-8285 (Q 03.08.05-13)
- Description of issue
  - Staff request to confirm that the crane loads are considered in the design of the entire containment and other seismic category I structures, and that the crane/trolley load is considered in different positions to maximize the loads on the structures.
- Resolution
  - The crane loads were considered in the design of the containment structure and other seismic category I structures. The positions to give maximum polar crane loads in the design of containment structures were described in DCD Subsection 3.8.1.3.2. In case of auxiliary building, the enveloped crane loads from various FHA crane locations were considered and described in DCD Subsection 3.8.4.3.1. The parking positions of the actual polar crane and trolley will be verified by COL 3.8(21).

Summary of Open Items

❖ Open Item: Subgrade Modulus of Soil Profiles for APR1400

- Related RAIs : 255-8285 (Q 03.08.05-16)
- Description of issue
  - Staff requested to update stability evaluations in all applicable parts of DCD Sections 3.8.5 and 3.8A, and technical report to reflect the current models, analysis approach, and results. Also Staff requested to provide the descriptions for bearing pressure evaluations in the EDGB and DFOT structures
- Resolution
  - The updated basemat stability evaluations with the application of the distributed soil spring elements for static conditions and the solid soil elements for dynamic conditions were added in DCD Subsection 3.8A.1.4.2.3.3 and Technical Report (APR1400-E-S-NR-14006, Stability Check for NI Common Basemat, Section 4.2.3 and Table A-7).
    The description for bearing pressure evaluation were added in DCD Subsections 3.8A.1.4.2.3.5 for NI common basemat and 3.8A.3.4.1 for EDGB and DFOT basemats, respectively.

Summary of Open Items

❖ Open Item: Differential Settlement of NI Common Basemat

- Related RAIs : 255-8285 (Q 03.08.05-17)
- Description of issue
  - Staff requested to explain the determination method of differential settlement.
- Resolution
  - The differential settlements were determined based on the construction phase and the post-construction phase, and a detailed procedure was added in the DCD Subsection 3.8.5.4.2 and Technical Report (APR1400-E-S-NR-14006, Stability Check for NI Common Basemat, Section 5).

Summary of Open Items

❖ Open Item: Pump-Induced Loads on RVI Components

- Related RAIs
  - RAI 151-8078 (Q 03.09.02-9)
- Description of issue
  - KHNP was requested to provide details on application of measured pump pressure pulsations and a related hydraulic model in order to calculate deterministic forcing function for loads on RVI components, and to clarify the application adequacy for design conservatism.
  - KHNP was requested to add a description of DPVIB computer program to DCD Section 3.9.1.
- Resolution:
  - KHNP provided details on application of measured pump pressure pulsations and a related hydraulic model to APR1400 design, and clarified the application adequacy.
  - KHNP submitted a description of DPVIB computer program was added to DCD Section 3.9.1.

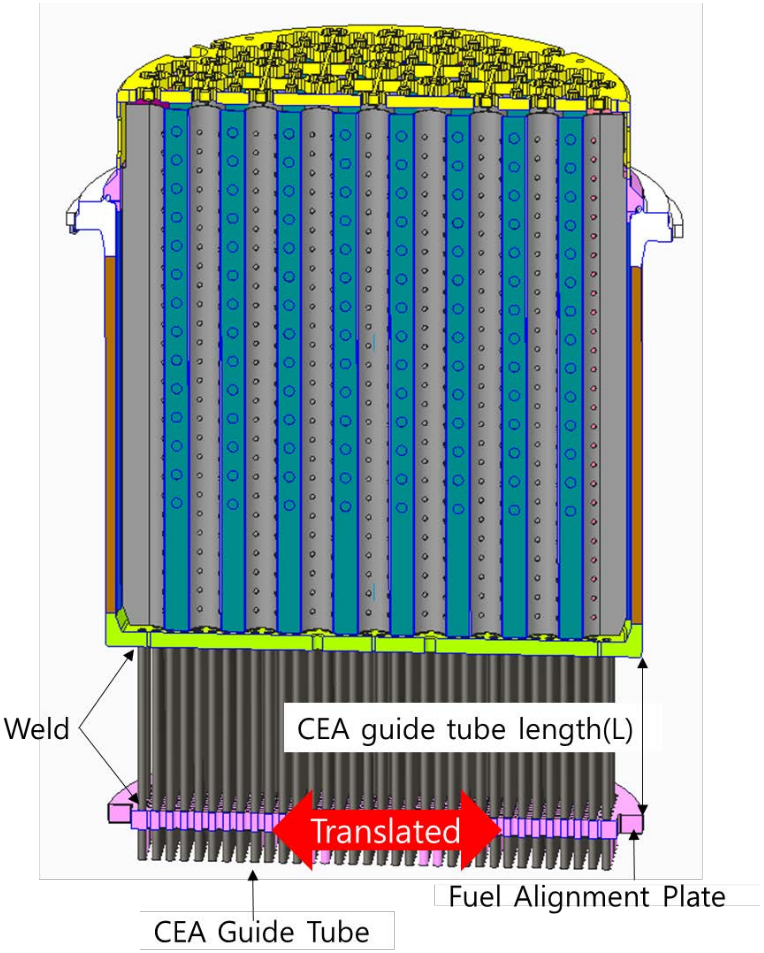
Summary of Open Items

❖ Open Item: Control Element Guide Tube buckling integrity and Operating experience

- Related RAIs
  - RAI 92-8068 (Q 03.09.05-6)
- Description of issue
  - The staff concerned regarding buckling integrity of the control element guide tube under Level D condition.
  - KHNP was requested to address any other operating experiences multiple domestic plants in Korea have similar design as APR1400 design.
- Resolution:
  - Theoretically, if the guide tube is deflected, it cannot buckle. Therefore, buckling does not occurred in the guide tube. But, the buckling integrity of CEA guide tube was evaluated and the detail calculation was submitted in third supplemental response.
  - KHNP reviewed the operational history the Korean domestic plants and confirm that there have been no damage in the fuel assembly.

Summary of Open Items

*** Buckling Analysis**



***Position requirement for CEA guide tube**

- One of functional requirements of CEA is that mechanical clearances of the CEAs within the fuel and reactor internals are such that the requirements for CEA positioning and a reactor trip are attained under the most adverse accumulation of tolerances.

- The alignment check provides reasonable assurance that the frictional force that could result from adverse tolerances is below the force that could significantly increase scram time.

- The surveillance requirement(SR 3.1.4.5) in TS, Section 3.1.4 describes "Verify each full strength CEA drop time from the fully withdrawn position to the 90% insertion position is below 4 seconds.

Summary of Open Items

Open Item: In-Core Instrumentation Support System

- Related RAIs
  - RAI 92-8068 (Q 03.09.05-9)
- Description of issue
  - KHNP was requested to provide information for the static O-ring seal, including its classification and design requirement. A summary of this information should be included in the DCD.
- Resolution:
  - KHNP provided explanation for the classification, design requirements, and design conditions of the static O-ring seals and the summary of these items were incorporated into the DCD Tier 2 (Rev. 1), Subsection 3.9.5.1.4.
  - KHNP provided test results of the static O-ring seal to ensure the reactor pressure boundary is maintained throughout its service life.

Summary of Open Items

Open Item: Loading Condition – Level D Service Loading

- Related RAIs
  - RAI 92-8068 (Q 03.09.05-16)
- Description of issue
  - KHNP is requested to describe and justify any differences between DCD Tier 2, Table 3.9-1 and the loads applied to the reactor pressure vessel internals for all service levels.
- Resolution:
  - The major difference between DCD Tier 2, Table 3.9-1 is that IRWST discharge loads are described in the loads and load combinations in the DCD, but Table 3.9-1 shows the only title of transient events.
  - KHNP provided the explanation for the events that result in an IRWST injection and discharge loads of the upset events and the faulted events listed in DCD Tier 2, Table 3.9-1.

Summary of Open Items

Open Item: Bases for Reactor Internals - Classification

- Related RAIs
  - RAI 92-8068 (03.09.05-18)
- Description of issue
  - KHNP was requested to provide all of reactor internals seismic classification.
  - KHNP was requested to provide complete classification list , DCD Tier 2 Table 3.2-1.
- Resolution:
  - KHNP provide revised response including explanation of all of the reactor internals are classified as seismic Category I and proposed revision to DCD Tier 2, Table 3.2-1.
  - The complete classification list , DCD Tier 2 Table 3.2-1 is incorporated into the DCD Rev.2.

Summary of Open Items

Open Item: Pump and Valve Inservice Testing Program Table

- Related RAIs
  - RAI 69-7994 (Q 03.09.06-9)

- Description of issue
  - The staff requested that KHNP to evaluate and modify all pumps and valve in Table 3.9-13 to be consistent with the NRC regulations in 10 CFR 50.55a, and guidance for IST.

- Resolution:
  - KHNP added additional information of inservice test requirements for pumps and valves in the RAI 69-7994 Q 03.09.06-9 (ML16242A438) including a proposed revision to DCD Tier 2, Table 3.9-13.

Summary of Open Items

❖ Open Item: Environmental Qualification Of Mechanical and Electrical Equipment

- Related RAIs : 27-7944 (Open item 03.11-1)
- Description of issue
  - KHNP was requested to provide justification why IEEE Std. 323-2003 is acceptable for qualification of Class 1E electrical equipment in the harsh environment.
- Resolution:
  - KHNP provided a justification for the use of IEEE Std. 323-2003, stating that it conforms with 10 CFR 50.49, that there are no technical differences between the 2003 and 1974 versions of the IEEE Std. 323, and that IEEE Std. 323-2003 reflects current practices for environmental qualifications. In addition, KHNP provided a basic table comparing the guidance contained in IEEE Std. 323-1974 and IEEE Std. 323-2003 by letter dated July 17. 2015(ADAMS Accession No. ML15198A260).

Summary of Open Items

❖ Open Item: Environmental Qualification of Mechanical and Electrical Equipment

- Related RAIs : 527-8686 (Open item 3.11-2)
- Description of issue
  - The staff requested KHNP to clarify that the qualification documentation will provide auditable records that show the equipment can perform its safety function during and following a DBE, as applicable.
  - The staff requested KHNP to discuss the other applicable DBE conditions and how the design is demonstrated to conform to 10 CFR 50.49.
  - The staff requested KHNP to clarify the discrepancy between IEEE Std.323-1974 and IEEE Std.323-2003 on the synergistic effects and explain how synergistic effects are considered for type test qualification in IEEE Std.323-2003.

Summary of Open Items

- Resolution:
  - KHNP provided a response that IEEE Std.323-2003 Section 4.4 states “The result of a qualification program shall be documented to demonstrate the equipment’s ability to perform its safety function(s) during its qualified life and applicable design basis events” which include the design event conditions such as high-energy line break, loss-of coolant accident, main steam line break, and/or safe shutdown seismic events, during and after which the equipment is required to perform its safety function, in Section 6.1.5.2 of IEEE Std.323-2003.
  - KHNP provided a response by justifying that the other accidents do not result in limiting conditions for safe shutdown consideration such as LOCA, MSLB and HELB; therefore, these are considered insignificant in the environmental qualification.
  - KHNP provided a response that IEEE Std.323-2003 Section 6.3.1.8.2, “Age conditioning,” refers to synergistic effects by stating “The sequence of age conditioning should consider sequential, simultaneous, and synergistic effects in order to achieve the worst state of degradation”. It is appropriate to mention in Section 6.3.1.8.2, “Age conditioning,” (a required part of type testing), because synergistic effects must be considered at the stage of age conditioning.

Summary of Open Items

❖ Open Item: Structural integrity of piping and pipe supports during the operation of SIT and FD

- Related RAIs : 549-8856 (Q 3.12-19)
- Description of issue
  - Staff requested to provide the followings:
    - ✓ Cavitation effects and vibration originating from the operation of the SIT and its FD in the structural design evaluation of the SIT, its discharge piping and pipe supports.
    - ✓ The operation of the SIT with its FD can result in other phenomena, such as water hammer.
    - ✓ The structural evaluation model of the SIT is coupled with the FD

Summary of Open Items

- Resolution:
  - KHNP provide the followings:
    - ✓ The cavitation noises are non-periodic and their frequencies are much higher than the modal frequencies of the SIT and the SI line, therefore, there exists no possibility of resonance due to the cavitation implosion.
    - ✓ Safety Injection System is designed to prevent the water hammer by eliminating possible void trap in the system based on NUREG/CR-6519. However, based on NRC feedback, the pressure and dynamic forces on the piping system due to SIT blowdown are calculated, which results that all pipe stress meet code allowable and the support can be acceptable in existing support design.
    - ✓ A structural evaluation model of the SIT for seismic analysis is coupled with the fluidic device.

Current Status

❖ **Chapter 3 is completed.**

- KHNP continues to monitor Chapter 3 to assure any conforming changes are addressed.
- 23 open items that were identified in Phase 2 and 3 have been resolved with adequate and sufficient discussion with the staff.

❖ **Changes in Chapter 3 as reviewed and marked-up in response to the RAIs were incorporated into the Rev.2 of the DCD.**

Attachment: Acronyms (1/2)

AB	Auxiliary Building
CFD	Computational Fluid Dynamics
COL	Combined License
CSDRS	Certified Seismic Design Response Spectra
DBE	Design Basis Event
DCD	Design Control Document
DFOT	Diesel Fuel Oil Tank
EDGB	Emergency Diesel Generator Building
FD	Fluid Device
HRHF	Hard Rock High Frequency
IEEE	Institute of Electrical and Electronics Engineers
IRWST	In-containment Refueling Water Storage Tank
ISRS	In-structure Response Spectra
ITAAC	Inspection, Test and Acceptance Criteria
KHNP	Korea Hydro and Nuclear Power Co.
LBB	Leak Before Break
NI	Nuclear Island

Attachment: Acronyms (2/2)

PICEP	Pipe Crack Evaluation Program
PRHA	Pipe Rupture Hazard Analysis
QA	Quality Assurance
RCB	Reactor Containment Building
RAI	Request for Additional Information
RG	Regulatory Guide
RVI	Reactor Vessel Internals
SIT	Safety Injection Tank
SRP BTP	Standard Review Plan Branch Technical Position
SRSS	Square Root of the Sum of the Squares
SSCs	Structures, Systems and Components
SSE	Safe Shutdown Earthquake
SSI	Soil-Structure Interaction
SSSI	Structure-Soil-Structure Interaction
V&V	Verification and Validation

Attachment : List of COL Item related to OIs (1/3)

COL Identifier	Description
COL 3.8(11)	The COL applicant is to verify that the coefficient of friction between the lean concrete and waterproofing membrane is greater than or equal to 0.55.
COL 3.8(18)	A detailed construction sequence analysis to determine the resulting construction settlements, including the various standard soils profiles (S01-S04, S06-S09) and sequencing of concrete pours for the NI common basemat (RCB and Auxiliary Building), and superstructure model (Auxiliary Building, internal structures, and Shell & Dome), is presented in Section 3.8.5.4.2. A comparison of the four types of construction settlements (i.e. maximum vertical settlement, tilting settlement, maximum differential settlement between structures, and angular distortion) to the maximum criteria listed in Table 2.0-1 is summarized in Tables 3.8-12 through 3.8-14, and Section 3.8.5.4.2.2.d. The COL applicant should use the construction sequence settlement analysis given in Section 3.8.5.4.2, substituting site-specific soil layer conditions, to ensure that the four types of settlement criteria described in Table 3.8-12 thru Table 3.8-14, and Section 3.8.5.4.2.2.d are satisfied. An alternative construction sequence and settlement analysis may be performed by the COL applicant in response to 1) the inability to meet the settlement criteria described in Table 3.8-12 thru Table 3.8-14, and Section 3.8.5.4.2.2.d using the DCD approach discussed in Section 3.8.5.4.2 or 2) Other site specific factors that may require a different construction plan and foundation sequence. However, in either case the COL applicant shall satisfy four types of settlement criteria described in Table 3.8-12 thru Table 3.8-14, and Section 3.8.5.4.2.2.d.

Attachment : List of COL Item related to OIs (2/3)

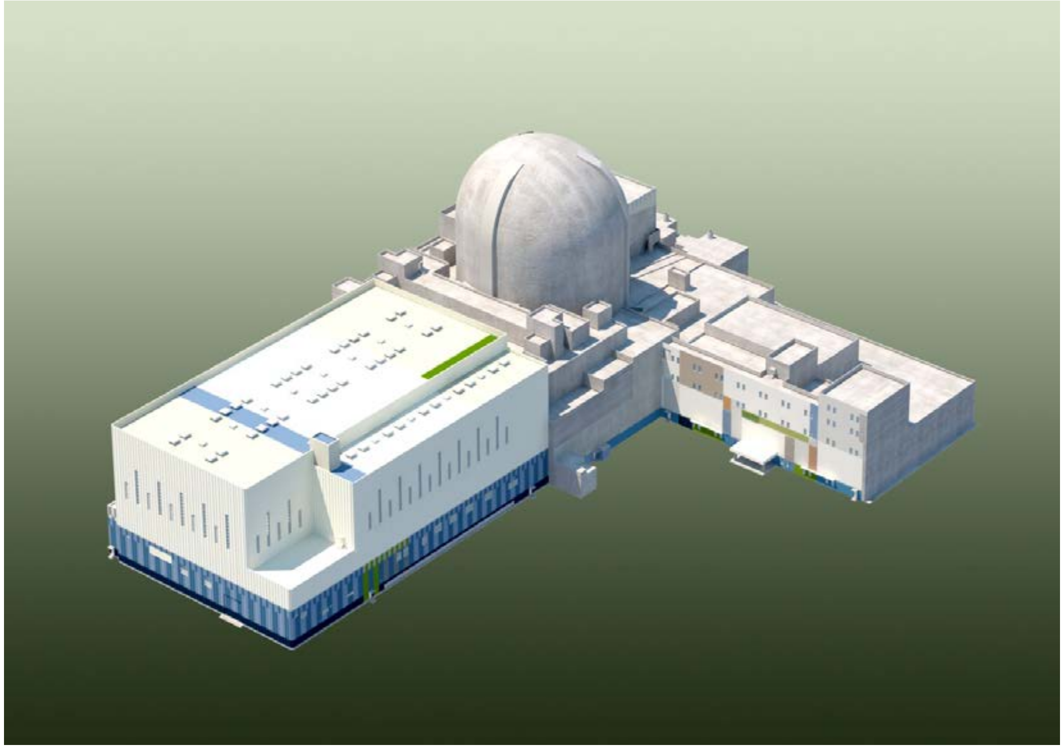
COL Identifier	Description
COL 3.8(19)	The following items should be considered by the COL applicant: 1) The surveyed soil profiles will be developed. 2) Based on the surveyed soil characteristics, differences from the DCD soil profiles may exist. These differences may include: a. Stiff or soft soil areas; b. Different soil types (e.g, cohesive); c. Potential for loss of cement in the mudmat; d. Non-uniformity of soil layers, or e. Other differences in the soil profile from the properties assumed in design certification. If any of these items and/or conditions are identified, then a site-specific evaluation[1)] shall be performed and checked for adequacy. 3) The time (i.e, short term and long term) instantaneous settlement and time-consolidation effects shall be evaluated in accordance with surveyed soil profiles regardless if a site-specific evaluation is needed under Item 2) above. The bearing pressure shall be checked to demonstrate acceptability with the acceptance criteria in DCD Table 2.0-1. Settlements shall be checked in Table 3.8-12 thru Table 3.8-14, and Section 3.8.5.4.2.2.d. 4) The COL applicant will build the seismic Category I structure according to the construction sequence used in the site-specific construction sequence analysis. 5) If a site-specific evaluation1) is required, the COL applicant should perform a construction sequence analysis based on the site-specific parameters. If the settlement including results of construction sequence analysis exceeds the acceptance criteria described in Table 3.8-12 through Table 3.8-14, and Section 3.8.5.4.2.2.d , the construction sequence will be modified to meet the acceptance criteria described in Table 3.8-12 through Table 3.8-14, and Section 3.8.5.4.2.2.d by COL applicant. 6) The effect on the design of seismic Category I structures due to construction sequence analysis shall be accounted for by the COL applicant. [1)] Evaluation includes basemat and superstructure design (forces/stresses), settlement evaluations, soil bearing pressure evaluation, and stability evaluation.

Attachment : List of COL Item related to OIs (3/3)

COL Identifier	Description
COL 3.8(20)	The COL applicant shall perform site-specific evaluations if the shear wave velocity is than the shear wave velocity profile used in the various basemat evaluations for design certification. The site-specific evaluations [settlement (maximum vertical displacement, tilt differential settlement between structures, angular distortion), soil bearing pressure (static and dynamic loading cases), overturning, and sliding] and 3D FEM global analysis for basemat design of seismic Category I structures shall be performed using the site-specific parameters (measured E_{static}, $E_{dynamic}$ consistent with soil strain assumed in SSI analysis) and the methodology described in DCD Tier 2, Subsection 3.8.5 and Technical report APR1400-E-S-NR-14006-P, Subsection 4.
COL 3.8(21)	The COL applicant is to confirm that the parking position of the crane and trolley when the crane is not being used is: location of polar crane: Az.280º, trolley location: 12ft 7in away from end of east part. The COL applicant is to confirm that this requirement is included in the technical specification of the COL application for the use of the polar crane.
COL 3.12(1)	If COL applicant finds it necessary to route ASME Class 1, 2, or 3 piping systems outside th e structure, the wind and/or tornado load must be included in the plant design basis loads c onsidering the site-specific loads.
COL 3.12(2)	The COL applicant will implement the monitoring program during the first preoperational tes ting and continue to monitor by using the fatigue monitoring system during the first cycle of operation to verify the design transients used in the structural design of the surge line.

NON-PROPRIETARY

APR1400 DCA
Chapter 2, Section 2.5: Geology, Seismology, and Geotechnical Engineering



KEPCO/KHNP

April 17, 2018

Contents

Overview of Section 2.5

Section	Title	Major Contents
2.5	Geology, Seismology, and Geotechnical Engineering	• Geologic • Seismologic • Geotechnical site parameters used for APR 1400 structural design and analysis

Overview of Section 2.5

❖ List of Submitted Documents

Document No.	Title	Revision	Type	ADAMS Accession No.
APR1400-E-S-NR-14006 -P & NP	Stability Check for NI Common Basemat	4	TER	-

❖ Summary of RAIs for Chapter 2

No. of Questions	No. of Responses	Not Responded	No. of OI
21	21	0	0

Summary of Issue Items

❖ Item: Static Stability of Foundation

- Description of issue

  Staff requested to provide bearing pressures and settlements of DCD Section 2.5.4 per the related open RAIs (RAI No. 255-8285, Q 03.08.05-7 and Q 03.08.05-16) of DCD Section 3.8.5 when they were resolved.

- Resolution

  - Based on the response of RAI for DCD Section 3.8.5, revised static/dynamic bearing pressures for NI Common Basemat, EDGB and DFOT were provided in the DCD Subsections 3.8A.1.4.2.3.5 and 3.8A.3.4.1. These bearing pressures will be verified by evaluating with the allowable bearing capacities based on the site-specific properties by COL 2.5(13).

  - Revised maximum differential settlements inside buildings and between buildings were provided in the DCD Table 3.8A-17, Table 3.8A-39 and DCD 3.8A.3.4.1. The predicted settlements will be verified whether they exceed the maximum settlement specified in DCD Table 2.0-1 for site suitability determination by COL 2.5(14).

Current Status

- **Section 2.5 is completed.**
  - KHNP continues to monitor Section 2.5 to assure any conforming changes are addressed.

- **Changes in Section 2.5 as reviewed and marked-up in response to the RAIs were incorporated into the Rev.2 of the DCD.**

Attachment: Acronyms

- COL/COLA: Combined License/Combined License Applicant
- DCD: Design Control Document
- DFOT: Diesel Fuel Oil Tank
- EDGB: Emergency Diesel Generator Building
- KEPCO: Korea Electric Power Corporation
- KHNP: Korea Hydro & Nuclear Power
- NI: Nuclear Island
- RAI: Request for Additional Information
- RCB: Reactor Containment Building
- RG: Regulatory Guide
- SSC: Structures, Systems, and Components

Attachment : List of COL Items

COL Identifier	Description
COL 2.5(13)	The COL applicant will evaluate the allowable bearing capacity of the subsurface based on the site-specific properties of the underlying materials, including appropriate laboratory test data to evaluate strength, and considering local site effects, such as fracture spacing, variability in properties, and evidence of shear zones. If the site-specific allowable bearing capacity is less than the maximum bearing demands specified in Table 2.0-1, a site-specific evaluation shall be performed by a COL applicant using the APR1400 basemat model and methodology described in Subsection 3.8.5.
COL 2.5(14)	The COL applicant will verify whether the predicted settlements within each building and between buildings exceed the maximum differential settlement specified in Table 2.0-1 for site suitability determination. If the predicted settlement exceeds the maximum value in Table 2.0-1, a detailed site specific evaluation shall be performed by a COL applicant to demonstrate acceptability. The COL applicant will also meet settlement criteria specified in COL 3.8(18) for construction sequence and post construction settlement limits.

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Presentation to the ACRS Subcommittee

Korea Electric Power Corporation
APR1400 Design Certification Application Review

Advanced Safety Evaluation Report

Chapter 3: Design of Structures, Systems, Components, and Equipment

April 17, 2018

Staff Review Team

United States Nuclear Regulatory Commission
Protecting People and the Environment

- **Technical Staff**
  - **Robert Roche-Rivera**, Structural Engineer, Structural Engineering Branch
  - **Vaughn Thomas**, Structural Engineer, Structural Engineering Branch
  - **Ata Istar**, Structural Engineer, Structural Engineering Branch

- **Project Managers**
  - **William Ward,** Lead Project Manager
  - **Tomeka Terry**, Chapter Project Manager

Seismic System Analysis SER Section 3.7.2

Robert Roche-Rivera

High Frequency Seismic Spectra – Seismic System Evaluation

- Seismic analysis and design of the APR 1400 standard plant are based on the CSDRS
  - ♦ Additionally, the APR1400 standard plant is evaluated for the effects of the HRHF spectra
- Original HRHF evaluation lacked technical justification for:
  - ♦ Number of spatial coherency modes used in the evaluation.
    - Insufficient, based on non-negligible difference in ISRS results from original alternative spatial coherency mode sets
  - ♦ ISRS spectral amplitude reductions greater than the limits set forth in SRP 3.7.2.II.4.
  - ♦ Lacked basis to demonstrate the qualification of the APR1400 structures to HRHF demands.

High Frequency Seismic Spectra – Seismic System Evaluation (Cont'd)

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- The applicant re-evaluated HRHF effects with consideration of more than twice the original number of spatial coherency modes.
  - Number of coherency modes shown to be adequate based on ISRS results
    - Close agreement between ISRS for alternative spatial coherency mode sets
  - Reductions to coherent ISRS found acceptable based on the applicant's consideration and demonstration of sufficient spatial modes in their HRHF evaluation.
  - APR1400 structures were demonstrated to be qualified for the HRHF spectra based:
    - Higher CSDRS spectral accelerations throughout dominant frequency range
    - Higher equivalent accelerations (used in the structural design) obtained from the CSDRS
    - Reinforcement (CSDRS demand) shown to be adequate for HRHF demand

Concrete and Steel Internal SER Section 3.8.3

Vaughn Thomas

Design of the Operating Concrete Floor Slabs

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- Reviewed APR1400 DCD Section 3.8A.1.4.3.1.3 in accordance with SRP 3.8.3 acceptance criteria
  - Description of the design and analysis procedures not provided
  - Analysis to demonstrate that the gap between the containment and the floor slabs was not provided
- Applicant performed additional calculation and sensitivity analysis of the concrete floor slabs following the decoupling criteria in SRP 3.7.2
- The results demonstrated that a gap of 2 1/16 in. is adequate to allow the relative displacements between the containment internal floors and the containment wall
- The applicant re-analyzed the CIS using the reaction forces obtained from local analysis of the concrete floor slabs and the design results showed an increase in member forces
- Provided markup copies of the DCD which describe in detail the design connection of the concrete slab and the steel beam between the containment wall and the SSW along with the new design results of the CIS

Foundations
SER Section 3.8.5

Ata Istar

Critical Sections

- In DCD Tier 2, Sections 3.8A.1.4, 3.8A.2.4, and 3.8A.3.4 “Analysis and Design for Critical Sections,” the applicant identified the critical sections for design of the RCB.
- The applicant did not capture all the critical sections following their criteria.
- The applicant added the missing critical sections in the DCD (structural steel and reinforced concrete sections).
- The staff determined that the design of these sections were consistent with SRP 3.8 and applicable codes and standards.

Waterproofing Membrane

- In APR1400 DCD, Section 3.8.5.1, “Description of the Foundations” no waterproofing membrane is identified for the foundation.
- The applicant responded that waterproofing membranes will be used for exterior below grade horizontal and vertical surfaces of structures in APR1400 design with a minimum coefficient of friction of 0.55.
- The applicant has assumed a coefficient of friction of 0.55 in the stability evaluation which is to be verified by COL 3.8(11).

Construction Sequence & Differential Settlements

- In APR1400 DCD, Section 3.8.5.1, “Description of the Foundations,” Section 3.8.5.4, “Design and Analysis Procedures,” Appendix 3.8A, “Structural Design Summary,” and TR APR1400-E-S-NR-14006, “Stability Check for NI Common Basemat,” no descriptions were provided for construction sequence and differential settlement of NI foundation.

- The applicant incorporated detailed explanations of the construction sequence, and clearly determined the criteria of four settlement types: (1) maximum vertical settlement, (2) tilt settlement, (3) differential settlement between buildings, and (4) angular distortion for the NI foundation.

NI Basemat Analysis

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- In DCD Tier 2, Section 3.8.5.4, “Design and Analysis Procedures,” it was not clear how seismic and other loads are determined, and applied to the various structures in the APR1400 basemat design.

- The applicant responded that for seismic loads the equivalent static analysis methodology is used, and the envelop of the results of the linear case (SRSS combination method) and the nonlinear case (100-40-40 combination method) for three soil profiles (soft, medium and stiff) are used in the analysis of NI basemat.

- The staff finds the response acceptable because the applicant used the equivalent static analysis methodology and combination methods for the design of the NI basemat consistent with the guidance provided in RG 1.92 and SRP 3.8.5.

Max. Bearing Pressure NI Basemat

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- TR APR1400-E-S-NR-14006, Section 3.2.5, “Applied Loads”
- The applicant was requested to determine the maximum soil bearing pressure for the NI basemat from linear and nonlinear dynamic load combinations based on the updated analyses.
- In its response, the applicant calculated the maximum soil bearing pressure for all load combinations (static, nonlinear, linear and SASSI analysis cases), and determined that they are lower than the allowable static and dynamic bearing pressure capacities.

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Presentation to the ACRS Subcommittee

Korea Hydro & Nuclear Power Co., Ltd (KHNP) and Korea Electric Power Corporation (KEPCO)

APR1400 Design Certification Application Review

Advanced Safety Evaluation Report

Chapter 3: Design of Structures, Systems, Components, and Equipment

April 18, 2018

Staff Review Team

United States Nuclear Regulatory Commission
Protecting People and the Environment

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  - **Alexander Tsirigotis,** Mechanical Engineer, Mechanical Engineering Branch
  - **Yuken Wong,** Senior Mechanical Engineer, Mechanical Engineering Branch
  - **Tom Scarbrough,** Senior Mechanical Engineer, Mechanical Engineering Branch
  - **Edward Stutzcage,** Health Physicist, Radiation Protection and Accident Consequences Branch
  - **Eric Reichelt,** Senior Materials Engineer, Materials and Chemical Branch
  - **Jorge Cintron- Rivera,** Electrical Engineer, Electrical Engineering Branch, NRR

- **Project Managers**
  - **Bill Ward,** Lead Project Manager
  - **Tomeka Terry**, Chapter Project Manager

Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping
SER Section 3.6.2

Renee Li

Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

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Review Approach

- The objective of the NRC staff's review of this section is to verify and to ensure that adequate protections have been provided against the effects of the postulated pipe ruptures and are in compliance with the GDC 4 requirement.
- Reviewed the applicant's criteria used to define the pipe break/leakage crack locations and configurations including the break exclusion area.
- Reviewed the applicant's methodology for addressing the dynamic jet impingement and blast wave effects in a Technical Report (TR), APR1400-E-N-NR-17001-P Revision 1.
- Reviewed the pipe break hazards analysis (PRHA) summary report, APR1400-E-N-NR-14004-P, Revision 2.

Staff Findings

- The applicant's criteria to define the pipe break/leakage crack locations and configurations including the break exclusion area were found acceptable.

Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

U.S.NRC
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- The TR methodologies in addressing the dynamic jet impingement and blast wave effects were found acceptable.
- The results of the PRHA summary report demonstrated that the APR1400 design is in compliance with the GDC 4 requirement and are acceptable.

Open Item Resolution

- Break Exclusion Area (Open Item RAI 41-7957, Question 03.06.02-2)

  Issue: The design provisions to be employed in the break exclusion area are consistent with the staff's guidance. However, the break exclusion areas are beyond the containment penetration area. The applicant was requested to justify the APR1400 DCD break exclusion area.

  Resolution: The applicant submitted a RAI response to justify the DCD break exclusion area for the APR1400 design.

  - The applicant explained how the DCD break exclusion design provisions were considered and applied to the results of the design of piping in the break exclusion area.
  - The APR1400 design ensures sufficient accessibility to perform a 100 percent volumetric inservice examination of the pipe weld within the break exclusion area.

Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

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- The results of the calculated maximum stress ranges were low compared to the relevant BTP 3-4 stress limit for postulating break locations.
- The RAI response included markups of Revision 1 to applicable subsections in DCD Tier 2 Section 3.6.2.

Open Item Closure: The NRC staff found that the applicant has adequately demonstrated its design provision and specifying a 100 percent volumetric inservice examination criteria meet the applicable BTP 3-4 break exclusion criteria in the NRC staff's guidelines. The applicant has appropriately justified the acceptability of expanding the break exclusion area beyond the containment penetration. In addition, the NRC staff found the proposed DCD markups included in the RAI response were acceptable. This item is being tracked as a Confirmatory Item.

Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

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- PRHA Summary Report (Open Item RAI 166-8198, Question 03.06.02-3)

  Issue: To support the NRC staff's safety determination on the APR1400 PRHA assessment, the applicant was requested to submit summary information on the PRHA results to demonstrate the APR 1400 design meet the GDC 4 requirement.

  Resolution: The applicant submitted a RAI response with an associated PRHA summary report, APR1400-E-N-NR-14004-P, Revision 2.

  - Described the criteria used in determining postulated pipe failure locations, and the methodology for assessing the jet impingement and pipe whipping effects.
  - Summarized the postulated break locations, the targets impacted by the respective break as well as their associates protection methods.

  Open Item Closure: The NRC staff found the PRHA report provided sufficient information to demonstrate that the PRHA were performed in accordance with the methodology and criteria described in the APR1400 DCD and were in conformance with the applicable SRP for which the NRC staff found acceptable.

  The results presented in the report demonstrated that the APR1400 design is in compliance with the GDC 4 requirement such that SSCs important to safety are designed to accommodate and protected against the effects of postulated pipe failures. This item was closed.

Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

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- Blast Wave and Potential Feedback Amplification and Resonance Effects (RAI 359-8448, Questions 03.06.02-6, -7, and -8)

  Issues: The applicant was requested to address the blast wave and potential feedback amplification and resonance affects as presented in a TR. The applicant indicated that it would submit a TR revision with alternative approaches to address these issues.

  Resolution: The applicant submitted TR, APR1400-E-N-NR-17001-P Revision 1 with alternative approaches to address these two issues.

  A. For the Blast Wave Effects:

  - Computational Fluid Dynamics (CFD) modeling was performed for some of the most limiting steam break locations.
  - CFD analysis approach was benchmarked against several experiments and analyses of similar conditions to verify its suitability.
  - The results of the APR1400-specific CFD analysis were used to develop a methodology for assessing the blast wave effects for other HELB locations that have not been analyzed by CFD. The methodology conservatively accounted for the initial energy available to form the blast wave and the effects of plant geometry on the blast wave loading on the impacted SSCs.

Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

B. For the Potential Feedback Amplification and Resonance Effects:

- In the TR, the applicant discussed how the multiple physical characteristics (e.g., lack of perpendicular flat surfaces of sufficient size to establish a feedback loop, irregularities in contours of the broken pipe end that distort the outgoing jet) of the APR1400 postulated HELB would prevent occurrence of a potential feedback amplification and resonance.
- The absence of HELB resonance effects was substantiated through a survey of experimental results.

<u>Open Item Closure:</u>

A. For Blast Wave Effects:

The NRC staff found that the applicant's CFD analysis performed for assessing the blast effects for the APR1400 plant were technically justified. The applicant has provided sufficient information to demonstrate the validity and applicability of the test data and methodology of the referenced open literature. The applicant has demonstrated that the methodology to account for effects of APR1400 plant geometry on the blast wave loads was conservative. The applicant has adequately addressed the NRC staff's concern on the blast wave effects as identified in Appendix A of SRP 3.6.2. This item was closed.

Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

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Open Item Closure:

B. For the Potential Feedback Amplification and Resonance Effects:

The NRC staff found that the applicant has adequately discussed how the multiple physical characteristics of the APR1400 postulated high-energy line breaks would prevent occurrence of a potential feedback amplification and resonance. The applicant also demonstrated that the absence of potential feedback amplification and resonance was substantiated by a survey of experimental results.

The NRC staff found the applicant's evaluation and approach to address potential feedback amplification and resonance effects as identified in Appendix A of SRP 3.6.2 acceptable because the applicant has adequately demonstrated reasonable assurance that this phenomenon is not a concern for the APR1400 plant. This item was closed.

Leak Before Break Evaluation Procedures SER Section 3.6.3

Eric Reichelt

Leak Before Break

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- Reviewed applicable APR1400 DCD sections in 3.6.3
- Reviewed DCD references for applicability and use
- Held public meetings with KHNP/KEPCO staff about technical issues and RAIs leading to proposed DCD markups
- The staff found these DCD sections acceptable
- The technical issues and response to RAIs by KHNP were acceptable and were therefore closed.
- RAI 525-8685, Question 03.06.09, was an open item discussed at previous ACRS meeting. The staff requested the applicant to review conflicting values between PICEP input and DCD, and to provide copies of the PICEP input files.

Leak Before Break

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- The staff reviewed the revised input values and calculations and received the PICEP code from the applicant.
- The staff performed a Confirmatory analysis of the applicant's Piping Evaluation Diagrams (PED) for the applicant's Surge Line, reactor coolant loop (RCL) piping (with the hot leg (HL) and cold leg (CL) evaluated separately), DVI-2 line, and shutdown cooling (SC) line (main run inside containment).

Leak Before Break

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- Based on the proprietary information provided, the staff was able to perform the confirmatory analysis of the LBB PED. The NRC staff concludes that there is reasonable assurance the applicant's LBB analysis bounds the normal operation and normal operation + safe shutdown earthquake design conditions and finds the applicant's LBB analysis acceptable.
- There are no open items for this section.

Leak Before Break

Overall Conclusions on LBB

- The staff evaluation concludes on a design specific and piping system specific basis that the acceptance criteria are satisfied, and, therefore, that dynamic effects of pipe rupture may be eliminated from design consideration.

Special Topics for Mechanical Components
SER Section 3.9.1

Tom Scarbrough

Special Topics for Mechanical Components

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- Design Transients: The APR1400 transient occurrences are conservatively designed, based on the certified System 80+ design transients.
- Computer Programs Used in Analyses: The NRC staff audited verification and validation (V&V) documents of the computer programs used in analyses.

  DPVIB computer program is used to calculate fluctuating pressure distribution in the down-comer region caused by RCP pressure pulsation. The staff found the output of the DPVIB is in agreement with the test data. The description of DPVIB computer program is added into DCD Tier 2.
- Evaluated the Faulted Conditions: The evaluation of Faulted Conditions is in conformance with ASME BPV Code, Section III, Appendix F.

ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures SER Section 3.9.3

Tom Scarbrough

ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

- Load combinations for ASME Code Class 1, 2 and 3 components and component supports conform to ASME BPV Code, Section III.
- Dynamic system and dynamic fluid loadings, respectively DF and DFL, explicitly are defined, and they conform to ASME BPV Code, Section III.
- Component supports are designed in accordance with ASME BPV Code, Section III, Subsection NF.

ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

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- Audit and follow-up audit conducted of component design specifications in accordance with 10 CFR 52.47 to establish that design criteria, analytical methods, and functional capability satisfy ASME Code requirements, and to confirm that design information from DCD is properly translated into design specifications.
- As result of audit, design specifications and DCD being updated.
- RAI issued to track completion of design specification and DCD changes.
- Confirmatory item to be closed when KHNP notifies NRC that design specifications and DCD have been updated to resolve audit follow-up items with appropriate NRC review.

Functional Design, Qualification, and IST Program
SER Section 3.9.6

Tom Scarbrough

Functional Design, Qualification, and IST Program

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- NRC staff evaluated APR1400 DCD Tier 2, Section 3.9.6, "Functional Design, Qualification, and Inservice Testing (IST) Programs for Pumps, Valves, and Dynamic Restraints," using SRP Section 3.9.6
- DCD provision for functional qualification of pumps, valves, and snubbers specifying ASME Standard QME-1-2007 as endorsed by Regulatory Guide 1.100 (Revision 3) is acceptable.
- NRC staff conducted audit of design specifications of APR1400 components in accordance with 10 CFR 52.47
- DCD description of IST Program based on ASME OM Code (2004 Edition through 2006 Addenda) as incorporated in 10 CFR 50.55a is acceptable for reference in COL application.
- DCD Revision 1 reviewed to verify that planned changes in RAI responses have been incorporated.

Functional Design, Qualification, and IST Program

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- RAI 69-2994, Question 03.09.06-1 Confirmatory Item: RAI responses with planned ITAAC changes are acceptable. Awaiting next DCD revision to verify changes have been incorporated.
- RAI 550-8737, Question 03.09.03-7 Confirmatory Item: Design Specification Follow-up Audit items for functional design and qualification are acceptable. Awaiting notification by KHNP that design specifications and DCD have been updated to resolve audit items.
- RAI 69-7994, Question 03.09.06-9 Closed: NRC staff reviewed DCD Revision 1 and confirmed that planned IST table changes have been incorporated.

Dynamic Testing and Analysis of SSCs SER Section 3.9.2

Yuken Wong

Dynamic Testing and Analysis of SSCs

- The staff reviewed the methodology, testing procedure, inspection program, and dynamic analyses conducted by KHNP to ensure the structural integrity and functionality of piping systems, mechanical equipment, and their supports under vibratory loading
- Specifically, reviewed six main areas:
  1. piping vibration, thermal expansion, and dynamic effects testing including the initial test program for ASME BPV Code, Section III, Class 1, 2 and 3 piping.
  2. seismic analysis and qualification of seismic Category I components
  3. dynamic system analysis for reactor internals under operational flow transients and steady-state conditions
  4. preoperational flow-induced vibration testing of reactor internals
  5. dynamic system analysis of the reactor internals under faulted conditions
  6. correlations of reactor internals vibration tests with the analytical results

Dynamic Testing and Analysis of SSCs

- Staff reviewed the comprehensive vibration assessment program (CVAP) report for APR1400 steam generator and reactor internals design in comparison with System 80 reactor design. The Palo Verde Unit I reactor is prototype and APR1400 is non-prototype Category 1.
- RAI 533-8718 Question 03-09-02-17 Closed: Applicant stated that the mechanical tanks will be designed for a ZPA of 50 Hz and the hydrodynamic forces exerted on the tank wall. A new COL item [COL 3.9(8)] was added to DCD for the COL applicant to perform the detailed analysis of mechanical tanks, including the effects of fluid sloshing.
- DPVIB computer code benchmarking: Applicant provided acceptable validation and verification information as documented in Section 3.9.1. Open item is closed.

Reactor Pressure Vessel Internals SER Section 3.9.5

Yuken Wong

Reactor Pressure Vessel Internals

- NRC staff evaluated the arrangement of reactor internals, their functions, flow path through the reactor vessel, and design criteria.
- Comparisons were made between the APR1400 reactor design and the CE System 80 reactor design, i.e. Palo Verde Units 1, 2 and 3, as the reactor designs are similar.
- Ensured the core support structures are constructed in accordance with ASME B&PV Code, Section III, Subsection NG. Internal structures are constructed in accordance with Subsection NG as guidelines.

Reactor Pressure Vessel Internals

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- RAI 92-8068, Question 03.09.05-6 Closed: KHNP investigated operating history of the Korean plants with control element guide tube design similar to that of APR1400 and confirmed that no failures have occurred on these components that would prevent control rod insertion.

- RAI 92-8068, Question 03.09.05-18 Confirmatory Item: KHNP provided clarification that all reactor internals (both core support structures and internal structures) are seismic Category I. Awaiting next DCD revision to verify changes have been incorporated.

Seismic & Dynamic Qualification of Equipment SER Section 3.10

Yuken Wong

Seismic & Dynamic Qualification of Equipment

- Reviewed DCD Tier 2, Section 3.10, 3.7B.7.4, and associated technical reports
- Verified equipment seismic qualification standards and methods
- Verified procedures to evaluate of effects of hard rock high frequency (HRHF) response spectra
- Conducted audit of procurement (design) specifications to verify seismic qualification methodologies
- RAI 550-8737, Question 03.09.03-7 Confirmatory Item: Design Specification Follow-up Audit items for seismic qualification are acceptable. Awaiting notification by KHNP that procurement specifications have been updated to address staff audit findings

Environmental Qualification of Mechanical and Electrical Equipment SER Section 3.11

Jorge A Cintron

Environmental Qualification of Mechanical and Electrical Equipment

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- Section 3.11 provides the APR1400 approach for environmental qualification of mechanical and electrical equipment.
- Staff reviewed the environmental qualification of mechanical and electrical equipment to verify the equipment is capable of performing its design functions under all normal environmental conditions, anticipated operational occurrences, and accident and post-accident environmental conditions. This equipment includes:
    - Safety-related equipment
    - Non-safety-related equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of specified safety functions
    - Certain post-accident monitoring equipment

Environmental Qualification of Mechanical and Electrical Equipment

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Confirmatory Item

- The staff evaluated applicant's response to RAIs requesting justification why IEEE Std. 323-2003 is acceptable for qualification of Class 1E electrical equipment in the harsh environment
- Regulatory Guide (RG) 1.89: provides an acceptable method for environmental qualification. Endorses IEEE Std. 323-1974, "IEEE Std. for Qualifying Class IE Equipment for Nuclear Power Generating Stations."
- The APR1400 deviates from RG 1.89 by using IEEE Std. 323-2003, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," for environmental qualification of mechanical and electrical equipment.

Environmental Qualification of Mechanical and Electrical Equipment

Confirmtory Item (Cont.)

- The staff evaluated the applicant's response to RAIs and concludes the applicant appropriately justified the technical differences between in the content of IEEE Std. 323-2003 and IEEE Std. 323-1974.
- Staff has completed evaluating the applicant revised response to the RAIs, therefore, this is a Confirmatory Item for Section 3.11

Radiological Equipment Qualification Review (Section 3.11)

Ed Stutzcage

Equipment Qualification (Radiological)

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Open Issues discussed at the previous ACRS Meeting (all associated with RAI 8089, Questions 03.11-9 and 03.11-11):

- Inconsistencies with Chapter 12 information regarding the neutron dose on the reactor Containment Building operating floor during normal operation
- Doses within the Auxiliary Building during accidents did not appear to adequately consider radiation streaming through containment penetrations.
- Additional information was needed regarding how some of the total integrated doses for areas were determined.
- Assumptions for post-accident fluid leakage rate outside of containment.

Equipment Qualification (Radiological)

Inconsistencies with Chapter 12 information regarding the neutron dose on the containment building operating floor during normal operation.

Issue:

- Additional information was needed to ensure that the total integrated neutron dose on the operating floor was adequate.

Resolution:

- Applicant updated technical report APR1400-E-X-14001-P to provide accurate operating floor dose information, based on the calculation data.
- Applicant also provided actual dose rate information on the operating floor for the operating APR1400 unit in Korea (Shin Kori Unit 3). The actual dose rate information was significantly lower than what is calculated and used in the equipment qualification (EQ) analysis for the operating area for the APR1400 DCA. Therefore, the EQ analysis is conservative.

Conclusion:

- As a result, the staff found the total integrated dose values for the operating floor to be acceptable.

Equipment Qualification (Radiological)

Doses within the Auxiliary Building during accidents did not appear to adequately consider radiation streaming through containment penetrations.

Issue:

- Doses in areas with penetrations were calculated at a dose point in the center of the room.

Resolution:

- Applicant re-analyzed the accident EQ total integrated doses for containment penetration areas and re-calculated the total integrated doses following an accident at a dose point 30 centimeters away from the penetrations. This provides an accurate estimate of the maximum doses that equipment within the room could be exposed to.

Conclusion:

- Therefore, the staff found the revised total integrated doses for penetration areas to be acceptable.

Equipment Qualification (Radiological)

Additional information was needed regarding how some of the total integrated doses for areas were determined.

Issue:

- Besides penetration areas (discussed on the previous slide), it was unclear how some other doses were calculated and there appeared to be potential errors with some information.

Resolution:

- Applicant revised the normal operation total integrated dose information for some areas:
  - Applicant provided more information in the technical report describing that the total integrated dose for areas without radiation sources were calculated assuming 1% failed fuel defects from adjacent sources, with additional margin added for conservatism. This is consistent with NRC guidance and is acceptable.
  - Applicant revised calculations for transfer tube inspection area to account for fuel transfer.
- Applicant provided additional updates clarifying the shielding thicknesses for certain areas, especially areas near and around demineralizers and filter areas.

Equipment Qualification (Radiological)

Additional information was needed regarding how some of the total integrated doses for areas were determined (continued).

Conclusion:

- The staff reviewed the shielding and dose rate information using MicroShield, as necessary and found them to be acceptable.

Equipment Qualification (Radiological)

Assumptions for post-accident fluid leakage rate outside of containment.

Issue:

- Applicant did not provide enough information explaining the allowed ESF leakage rate assumed in the EQ analysis and why the leakage was different than that used in the accident analysis.

Resolution:

- Applicant clarified that the allowed ESF leakage rate for EQ of 0.285 cubic feet per hour was based on the assumed leakage rates from valves and pumps specified in the response. The allowed leakage rate is doubled in accordance with RG 1.183. The applicant also proposed to update DCD Subsection 15.6.5.5.1.2 and Appendix B of APR1400-E-X-NR-14001-P to clarify that while the EQ analysis assumed 0.285 cubic feet per hour (which is doubled), the DCD Chapter 15 accident analysis used a different assumed leak rate of 18.9 L/hour (0.667 cubic feet per hour) doubled to 37.8 L/hour (1.335 cubic feet per hour).

Equipment Qualification (Radiological)

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Assumptions for post-accident fluid leakage rate outside of containment (continued).

Conclusion:

- The staff found the ESF leakage values used in the EQ analysis to be acceptable because the 0.285 cubic feet per hour is based on conservative leakage assumptions and it is the maximum leakage allowed. If the value was to be exceeded during plant operation, the licensee would have to evaluate the impacts on the equipment and take appropriate action.

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ASME Code Class 1, 2, and 3 Piping Systems and Associated Supports Design SER Section 3.12

Alexander Tsirigotis

Piping Analysis and Supports

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- Confirmed APR1400 piping and supports analysis incorporates NRC guidance and is in accordance with ASME BPV Code Subsections NB, NC, ND and NF as incorporated by reference in 10 CFR 50.55a.
- Conducted audit of piping stress analysis and support designs to confirm consistency with DCD.
- Held public meetings with the applicant to discuss technical issues during audits and RAIs leading to proposed DCD markups, revisions to technical reports and added piping analyses.
- The applicant has adequately addressed all NRC staff comments.

Piping Analysis and Supports

- During Phase 4 the applicant addressed structural effects on piping and its supports due to potential fluid dynamic loads occurring from the operation of the safety injection tank and its fluidic device.
- The applicant derived water hammer loads, which, when incorporated in the load combinations for pipe stress analysis and pipe support design, showed that piping and supports remained structurally adequate and met ASME Section III acceptance criteria
- The applicant modified the Safety Injection Tank Subsystem Test, DCD Tier 2 Section 14.2.12.1.22, to instrument the safety injection line to gather vibrational data and to ensure that vibration levels are within acceptable limits found in ASME OM-SG Part 3.

Presentation to the ACRS Subcommittee

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

Safety Evaluation with No Open Items:

Chapter 2 SITE CHARACTERISTICS

Sections 2.5

APRIL 17, 2018

Staff Review Team

- **Technical Staff**
  - Luissette Candelario, David Heeszel, Ricardo Rodriguez
- **Project Managers**
  - Bill Ward – Lead Project Manager
  - Carolyn Lauron – Project Manager

Technical Topics

Chapter 2 – Site Characteristics

- The APR1400 design assumes a site envelope that describes the geography and demography, nearby facilities, and postulated site parameters for the design, including meteorology, hydrology, geology, seismology, and geotechnical parameters.

- In November 2017, the staff presented the site characteristics related to:
  - Geography and demography
  - Meteorology
  - Hydrology

- Today's presentation covers the site characteristics related to geology, seismology and geotechnical engineering.

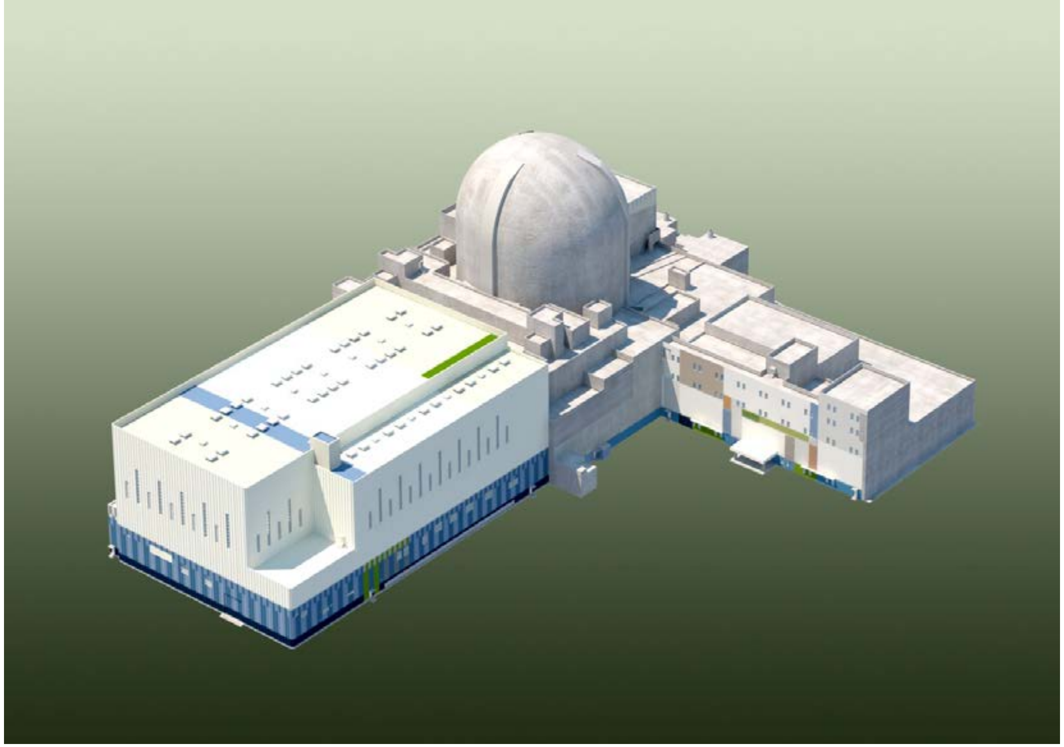
Technical Topics - Overview Chapter 2, Site Characteristics Section 2.5

- There were no open items previously identified.

- The applicant properly identified appropriate geologic, seismologic and geotechnical site parameters and properly identified the site specific information to be provided as part of any future COL application.

- We are currently reviewing DCD Revision 2 to ensure all confirmatory items have been incorporated.

APR1400 DCA
Chapter 15:Transient and Accident Analyses



KEPCO/KHNP

April 17, 2018

Contents

Overview of Chapter 15

Section	Title
15.0	Introduction : Transient and Accident Analyses
15.1	Increase in Heat Removal by the Secondary System
15.2	Decrease in Heat Removal by the Secondary System
15.3	Decrease in Reactor Coolant System Flow Rate
15.4	Reactivity and Power Distribution Anomalies
15.5	Increase in Reactor Coolant Inventory
15.6	Decrease in Reactor Coolant Inventory
15.7	Radioactive Material Release from a Subsystem or Component
15.8	Anticipated Transient without Scram
15A	Analytical Model for Determining Radiological Consequences of Accidents

Overview of Chapter 15

❖ List of Submitted Documents

Document No.	Title	Revision	Type	ADAMS Accession No.
APR1400-K-X-FS-14002-P P & NP	APR1400 Design Control Document Tier 2: Chapter 15 Transient and Accident Analyses	2	DCD	-
APR1400-Z-A-NR-14006-P	Non-LOCA Safety Analysis Methodology	1	TeR	ML17094A190
APR1400-F-A-NR-14003-P & NP	Post-LOCA Long Term Cooling Evaluation Model	1	TeR	ML17206A154
APR1400-F-A-NR-14001-P & NP	Small Break LOCA Evaluation Model	1	TeR	ML17114A523
APR1400-F-A-TR-12004-P & NP	Realistic Evaluation Methodology for LBLOCA of the APR1400	1	ToR	ML17265A104

❖ Summary of Open Items

* 10 open items at phase 3
* 10 open items are closed in phase 4

Overview of Chapter 15

❖ List of Open Items

No.	Related RAI	Topic	ADAMS Accession #
1	RAI 108-7973 (Q 15.00.03-1) & RAI 368-8470 (Q 14.03.08-14)	Periodic Reopening of the MCR Air Intakes	ML15344A121 ML17257A542
2	RAI 17-7917 (Q 15.04.06-1)	Complete Mixing Model	ML15146A260
3	RAI 511-8668 (Q 15.04.06-8)	VOPT During Boron Dilution at Power	ML16232A625
4	RAI 143-8092 (Q 15.06.05-1)	Loop Seal Clearing and Reformation	ML16363A031
5	RAI 318-8337 (Q 15.06.05-2)	SBLOCA Dense Break Spectrum Analysis	ML16194A269
6	RAI 398-8457 (Q 15.06.05-5)	Boron Precipitation	ML16363A415
7	RAI 404-8488 (Q15.06.05-10)	Upper Bound of SBLOCA Break Size	ML16201A284
8	RAI 404-8488 (Q15.06.05-11)	LOCA Deposition Model (DM)	ML16202A506
9	RAI 430-8455 (Q 15.06.05-22)	Boron Dilution in SBLOCA	ML16363A035
10	RAI 5-7954 (Q 11, 12, 16 & 18) & RAI 399-8510 (Q 15.06.05-7)	Impacts of Thermal Conductivity Degradation LBLOCA DCD Revision	ML17335A069 ML17223A687

Summary of Open Items

❖ Open Item: Periodic Reopening of the MCR Air Intakes

- Related RAIs : 108-7973 (Q.15.00.03-1)
- Open item reference: RAI 368-8470, Q 14.03.08-14
- Description of issue
  - The MCR ventilation system design provides for automatic selection of the least contaminated outside air intake. And the closed MCR intake will re-open automatically so that the intake with the higher radioactive can be selected again and closed.
  - Staff required to show that the increase in dose from both air intakes being open periodically can be encompassed by the current design margin, and DCD update.
- Resolution
  - KHNP provided the additional information on the dose evaluation including the increase of the dose can be compensated from the design margin of 20% (KHNP used the reduction factor of 8 instead of 10 for the MCR intakes x/Qs)
  - DCD is revised.

Summary of Open Items

❖ Open Item: Complete Mixing Model

- Related RAIs : 17-7917(Q 15.04.06-1)
- Description of issue
  - NRC Staff noted that the flow rate in Modes 4 and 5 when all RCPs are idle and with one shutdown cooling train in service may not be sufficient to assume complete RCS mixing.
  - Staff requested to provide the followings:
    - ✓ Justification that the complete mixing model is conservative including any potential effects of incomplete lower plenum mixing.
    - ✓ Demonstrate that the source range detectors can sense any postulated incomplete mixing.
- Resolution:
  - KHNP decided to add additional valve which is classified as Quality Group C to block the flow path from the unborated water source in Modes 4 and 5 with all RCPs idle.
  - In addition, KHNP added new TS 3.1.12 and its associated Bases to prohibit the boron dilution operation, and eliminated the quantitative analysis for these Modes in DCD 15.4.6.

Summary of Open Items

❖ Open Item: VOPT During Boron Dilution at Power

- Related RAIs : 511-8668(Q 15.04.06-8)
- Description of issue
  - NRC Staff noted that a slow reactivity insertion transient such as a boron dilution at power would not be protected by the variable overpower trip.
  - Staff requested to provide the followings:
    - ✓ Which reactor trip would occur for an at power boron dilution event.
- Resolution:
  - KHNP provided the clarification that the core protection calculator system (CPCS) DNBR trip or CPCS auxiliary trips can protect the DNBR limit during a slow reactivity insertion event.

Summary of Open Items

❖ Open Item: SBLOCA Dense Break Spectrum Analysis

- Related RAIs : 318-8337 (Q 15.06.05-2)
- Description of issue
  - Provide the results of a finer break spectrum for both the DVI line and PDL breaks
- Resolution:
  - KHNP performed the dense break spectrum analysis for the DVI line and PDL breaks.
  - Break spectrum analysis number is increased from 10 to 37
  - The limiting DVI line break size is changed from 0.4 ft^2 to 0.1364 ft^2
  - LOCA criteria is still satisfied. (DCD Table 15.06.05-10)
  - KHNP revised the RAI response, TeR (APR1400-F-A-NR-14001-P) and DCD chapter 15.6.5.

Summary of Open Items

❖ Open Item: Upper Bound on the SBLOCA Break Size

- Related RAIs : 404-8488 (Q 15.06.05-10)
- Description of issue
  - Demonstrate that the 0.5 ft^2 break size remains the largest SBLOCA which must be considered.
- Resolution:
  - KHNP responded that the analysis with the breaks larger than 0.5 ft^2 are treated by the current LBLOCA methodology.
  - KHNP presented results comparing the LBLOCA and SBLOCA PCTs.
  - The PCT calculated by the LBLOCA methodology was greater than the PCT calculated by the SBLOCA methodology.
  - No change in DCD.

Summary of Open Items

❖ Open Item: Thermal Conductivity Degradation (TCD)

- Related RAIs : 5-7954 (Q 11, PLUS7 Fuel Design for the APR1400)
- Description of issue
  - NRC Staff noted that FATES3B does not account for the effect of TCD and requested a TCD impact on fuel design and safety analyses.
- Resolution:
  - TCD penalty was determined based on the analysis of comparison to the experimental data at various burnups , and KHNP performed the analyses for the design evaluations and the related safety areas with TCD penalty.
  - For CEA ejection, peak radial average fuel enthalpy is increased less than 11 cal/g and peak centerline temperature is increased about 317 ^{o}F at HFP case. But Peak radial average fuel enthalpy is much lower than 230 cal/g, and no fuel centerline melting occurs.
  - For LBLOCA, the details of TCD effect will be discussed in LBLOCA Topical Report section.
  - TCD-affected DCD Tier 2 chapters and LBLOCA Topical Report were revised.
  - All the RAIs including impact of TCD have been resolved.

Summary of Open Items

❖ Open Item: Loop Seal Clearing and Reformation

- Related RAIs : 143-8092 (Q 15.06.05-1)
  - The details will be explained in future LTCC section.

❖ Open Item: Boron Precipitation

- Related RAIs : 398-8457 (Q 15.06.05-5)
  - The details will be explained in future LTCC section.

❖ Open Item: LOCA Deposition Model

- Related RAIs : 404-8488 (Q 15.06.05-11)
  - The details will be explained in future LTCC section.

❖ Open Item: Boron Dilution

- Related RAIs : 430-8455 (Q 15.06.05-22)
  - The details will be explained in future LTCC section.

Current Status

- ❖ **Chapter 15 is completed.**
  - KHNP continues to monitor Chapter 15 to assure any conforming changes are addressed.
  - 10 open items, that were identified in Phase 3, have been resolved with adequate and sufficient discussion with the staff.

- ❖ **Changes in Chapter 15 as reviewed and marked-up in response to the RAIs is incorporated into the revision (Rev.2) of the DCD.**

Acronyms

CEA	Control Element Assembly
CPCS	Core Protection Calculator System
DVI	Direct Vessel Injection
LOCA	Loss of Coolant Accident
LTC	Long Term Cooling
LTCC	Long Term Core Cooling
MCR	Main Control Room
PCT	Peak Cladding Temperature
PDL	Pump Discharge Leg
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
TCD	Thermal Conductivity Degradation
VOPT	Variable Over Power Trip

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Presentation to the ACRS Subcommittee

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

Safety Evaluation with No Open Items: Chapter 15
Transient and Accident Analyses

By

Timothy Drzewiecki, Michelle Hart, Shanlai Lu

April 17, 2018

Open Item 15.0.3-1

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Modeling of Control Room Emergency Makeup Air Cleaning System Operation

Description: (RAI Question 14.03.08-14 sub-question 6.b)

- DCD did not clearly describe system operation where control logic automatically re-opens closed CR air intake isolation dampers at a preset interval for a short time to determine which intake has lower radioactivity concentration. The COL applicant is to choose both the interval time between damper re-openings and the length of time the intakes are both open (COL item 9.4(2))

Closure: (confirmed incorporated in DCD Rev. 2)

- RAI response described the modeling assumptions for damper operation and described the level of conservatism and margin in the control room dose analyses
- Revisions made to DCD Sections 9.4.1 and 15.0.3 to describe more clearly the system operation and dose analysis assumptions
- ITAAC revised to clarify damper operation
- COL item 9.4(2) revised and COL items 15.0(2) and 15.0(3) added to identify that the COL applicant verify that the damper re-opening interval and duration are accounted for in the dose analyses

Open Item: 4.2-1 in Chapter 15

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Fuel Pellet Thermal Conductivity Degradation

Description

- Staff SER Section 4.2 “Fuel Design” identified this issue as an Open Item for fuel performance evaluation due to FATES3B code deficiency
- As a result, the initial steady state fuel center line temperature and the total core sensible heat/stored energy have been underestimated

Closure

- Fuel performance CESEC-III analyses are not impacted:
  - Analyses used conservative values for gap conductance that bound any increase in fuel temperature associated with TCD
  - Large uncertainty accounted for in the fuel temperature coefficient bounds the TCD impact on the reactivity feedback during a transient.

Open Item: 4.2-1 in Chapter 15

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Fuel Pellet Thermal Conductivity Degradation

Closure (Cont.)

- CETOP thermal margin analyses not impacted because these analyses use "dummy" rod (i.e., no modeling of heat transfer through the fuel pin, only deposit heat flux at fuel surface)
- DCD Section 15.4.8, CEA Ejection, fuel rod enthalpy re-evaluated (analysis still performed with STRIKIN-II):
  - Pellet thermal-conductivity multiplier applied in STRIKIN-II to increase fuel temperature to match TCD penalized FATES3B value.
  - Results still showed no fuel failures associated with high cladding temperature (at HZP) or PCMI. All fuel failures attributed to DNBR (not affected by TCD).
  - CEA ejection reanalysis results satisfied all criteria of against SRP 4.2 Appendix B and DG-1327
- DCD Section 15.6.5, LBLOCA, TCD penalty incorporated into LBLOCA methodology (APR1400-F-A-TR-12004)

Therefore, all the Chapter 15 re-analyses regarding TCD are acceptable

RAI Questions 15.6.5-1, 2

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Pending Large Break LOCA Topical Report Review

Description:

RAI Questions 15.6.5-1, 2 were created to track the pending review of the LBLOCA topical report and the final LBLOCA analysis results

Closure:

Details will be discussed during the LBLOCA topical report presentation

Open Item 15.4.6-1

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Inadvertent Boron Dilution During Modes 4 and 5

Description:

RAI 7917 was issued to request justification for the use of a complete mixing model in the Boron dilution analysis for Modes 4 and 5. Staff was concerned about the possibility of reactor core returning to critical as a result of inadvertent fresh water addition

Closure:

KHNP responded to the RAI with hardware design changes and relevant technical specification revisions

- New isolation valve (CV-186) added in the reactor makeup water line to block the flow path that could allow unborated water to reach the RCS during Modes 4 and 5
- New TS 3.1.12 added and revised TS 3.9.7 with relevant DCD changes

The design change will prevent the inadvertent fresh water injection by isolating fresh water source using the new isolation valve following the revised technical specifications. Staff considers the design changes and relevant technical specification modifications acceptable

Open Item 15.4.6-2

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Variable Overpower Trip During Boron Dilution at Power

Description:

RAI 511-8668 Question 15.4.6-8 was issued by the staff to request information about the DNBR protection provided by the variable overpower trip during a slow reactivity insertion transient, such as a boron dilution at power

Closure:

KHNP responded to the RAI and indicated that variable overpower trip would not be appropriate to prevent the DNB violation during such an event. The core protection calculator system (CPCS) DNBR trip or CPCS auxiliary trips are credited to protect the DNBR limit. Staff reviewed the RAI response and relevant analyses. It is concluded that DNBR limit is protected during a slow reactivity insertion event

Open Item 15.6.5-4

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SBLOCA Upper Bound Break Size and PCT

Description:

Staff issued RAI 8488 Question 15.06.05-10 and supplemental RAI request to ensure the full coverage of the entire break spectrum by both SBLOCA and LBLOCA methodology

Closure:

KHNP submitted the response and relevant analyses indicating that both LBLOCA and SBLOCA methodology predicted the PCT less than 2200 ˚F for a break size of 0.5 ft^2. Therefore, the results are acceptable

Open Item 15.6.5-5

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Boron Precipitation During Long Term Cooling

Description:

Staff was concerned that the mixing volume assumed by KHNP for the region above the core could be non-conservative and the switch-over time during a cold leg break LOCA event could be too long

Closure:

KHNP re-analyzed the Boron precipitation phenomenon and submitted all the revised analysis results and relevant DCD changes. In particular, the time of switch-over from DVI line to hot leg has been reduced from three hours to two hours. The design change is considered acceptable

Open Item 15.6.5-6

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Boron Dilution During LOCA Long Term Cooling Phase

Description:

- RAI 8455 Question 15.06.05-22 was issued based on Reg. Guide 1.206 regarding GSI-185 to request specific analysis for APR1400 conditions
- The phenomenon is possible due to the presence of a loop seal

Closure:

- KHNP responded to the RAI and performed APR1400 specific analysis
- Conservative assumptions were introduced in the calculation to determine the loop seal fresh water slug discharge based on PKL tests
- The calculated Boron concentration in the core has sufficient margin to maintain sub-criticality during post LOCA long term cooling phase. Therefore, the shutdown margin is maintained

Open Item 15.6.5-7

Non-proprietary

LOCA Deposition Model

Description:

Staff issued RAI 404-8488 Question No. 15.06.05-11 to obtain additional information about RCS coolant temperature, latent fiber weight percent and other relevant information as the input to the LOCA deposition model

Closure:

Based on the additional information submitted by KHNP, staff concluded that KHNP applied the approved Westinghouse LOCA deposition model properly and the calculated peak cladding temperature is less than 800 °F. Therefore, this open item is closed

Open Item 15.6.5-8

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Loop Seal Clearing During SBLOCA and Long Term Cooling

Description:

As the result of a loop seal, the core two-phase water level could be depressed due to the loop seal formation and clearing. Staff issued RAIs regarding the potential impact on peak cladding temperature during a SBLOCA and its long term cooling phase.

Closure:

KHNP reanalyzed the SBLOCA event considering the existence of the loop seal and demonstrated that although the heat-up is predicted, the impact on PCT is not significant. Staff reviewed the analysis methodology and concluded that the method is conservative. The results show that PCT is less than 2200 ˚F during the blowdown phase and less than 800 ˚F during the long term cooling phase.

Conclusion:

NRC staff reviewed all the supplemental information provided by KHNP as the result of Phase IV review and found it to be acceptable