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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

+ + + + +

MONDAY,

JUNE 5, 2017

+ + + + +

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

JOSE MARCH-LEUBA, Member

DANA A. POWERS, Member

JOY REMPE, Member

PETER RICCARDELLA, Member

GORDON R. SKILLMAN, Member

JOHN W. STETKAR, Member

DESIGNATED FEDERAL OFFICIAL:

CHRISTOPHER BROWN

ALSO PRESENT:

TONY AHN, KHNP

BUMSIK CHANG, KEPCO E&C

HOONIN CHO, KEPCO E&C

BOSUNG CHOI, KEPCO E&C

JORGE CINTRON-RIVERA, NRO

CARL CONSTANTINO, BNL

NICHOLAS HANSING, NRO

DAVID HEESZEL, NRO

JOHN HONCHARIK, NRO

TOM HOUSTON, BNL

JASON HUANA, NRO

SEOKHWAN HUR, KEPCO E&C

ATA ISTAR, NRO

HYEOK JEONG, KHNP

JIN JUNG, KHNP

DAEJEONG KIM, KEPCO E&C

JUNGHO KIM, KHNP

TAE HAN KIM, KEPCO E&C

YOUNGHOON KIM, KEPCO E&C

YUNHO KIM, KHNP

SUNGUK KWON, KHNP

YOUNGMAW KWAN, KEPCO E&C

YIU LAW, NRO

TUAN LE, NRO

JONG BO LEE, KHNP

KYUMIN LEE, KEPCO E&C

SANGGYU LEE, KHNP

YONGSUN LEE, KHNP and KEPCO E&C

CHANG-YANG LI, NRO

RENEE LI, NRO

DAEHEON LIM, KEPCO E&C

KWANGIL LIM, KEPCO E&C

SHANLAI LU, NRO

TIM LUPOLD, NRO

JOHN MA, NRO

STEVE MANNON, AECOM

TANIA MARTINEZ-NAVEDO, NRR

MIKE MCCOPPIN, NRO

JINSUO NIE, NRO

RYAN NOLAN, NRO

JIYONG OH, KHNP

CHONGHO PARK, KEPCO E&C

HONGSUN PARK, KHNP and KEPCO E&C

JAEWAN PARK, KHNP

SUNWOO PARK, NRO

SHEILA RAY, NRR

ERIC REICHELT, NRO

ROBERT ROCHE-RIVERA, NRO

TOM SCARBROUGH, NRO

ROBERT SISK, Westinghouse

EDWARD STUTZCAGE, NRO

KIKWANG SUNG, KEPCO E&C

TOMEKA TERRY, NRO

VAUGHN THOMAS, NRO

ALEXANDER TSIRIGOTIS, NRO

MARIELIZ VERA, NRO

DAVE WAGNER, AECOM

JAY WALLACE, RES

BILL WARD, NRO

DAN WIDREVITZ, NRO

YUKEN WONG, NRO

JOHN WU, NRO

JIM XU, NRO

JINKYOO YOON, KHNP and KEPCO E&C

KISEOK YOON, KEPCO E&C

SUNGHO YOU, KEPCO E&C

KWANHEE YUN, 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. The ACRS members in attendance are Dick Skillman, Dana Powers will be here shortly, John Stetkar, Jose March-Leuba and Joy Rempe.

The purpose of today's meeting is for the Subcommittee to receive briefings from Korea Electric Power Corporation, KEPCO, and Korea Hydro and Nuclear Power Company, Limited, KHNP, regarding their design certification application and the NRC staff regarding their safety evaluation with open items, specific of Chapter 3, Design of Structure Systems and Components and Equipment.

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.

An announcement is published in the Federal Register.

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

Written comments are also welcome.

I might add, also, that Chris Brown, who's here now, is the Designated Federal Official for this meeting.

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

The rules for participation in today's meeting were announced in the Federal Register on Tuesday, May 30th, 2017. The meeting was announced as an open/closed public meeting. This meant that the Chairman can close the meeting as needed to protect information proprietary to KHNP or its vendors.

I believe that this is all open today.

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

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

Participants should first identify themselves and speak with sufficient clarity and volume so that they can be readily heard. And, we keep going through this for every meeting, it seems. It's like a social experiment but the microphones that are in front up here, there's a little light on the top, that's not what you press to speak. There's a little button on the bottom.

We have a bridge line established for interested members of the public to listen in. The bridge number and password were published in the agenda posted on the NRC public website.

To minimize disturbance, this public line will be kept in a listen only mode. The public will have an opportunity to make a statement or provide comments at a designated time towards the end of this meeting.

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

And, we have been joined by the inestimable Peter Riccardella.

And by Bill Ward, NRO Project Manager to introduce the presenters and start the briefing.

Bill?

MR. WARD: Thank you.

The staff and KHNP are pleased to present one more chapter. This is the second to the last meeting. We'll be done with the presentations to the Subcommittee for Phase II versions from the SCE.

I know that it's a long chapter and it's only one day. So, I wanted to point out that the staff presentation will only cover certain sections where there was significant issues and they'll point that out later in their presentation.

So, we will take questions on any section that's not specifically mentioned in the discussion, but we look forward to being able to present that.

And, that's all I have today.

Thank you.

CHAIRMAN BALLINGER: The floor is yours.

MR. SISK: Thank you.

This is Bob Sisk, let me introduce Mr. Yunho Kim from KHNP.

MR. Y. KIM: Yes, I am Yunho Kim from

KHNP. I'm the APR1400 Project Manager. So, I have a -- at this time, we have the meeting until now. So, I look forward to having a good discussion today.

I am turn the presentation to Bob and to Mr. Yoon.

Thank you.

MR. SISK: Mr. Yoon?

MR. YOON: Good morning, ladies and gentlemen. My name is Jinkyoo Yoon, working for Nuclear Engineering Department at KEPCO Engineering and Construction.

This morning, I will -- I'm going to talk about design of structures, system, component and equipment of the APR1400.

And so, I'm very pleased to have this opportunity to present the overview, outlines of Chapter 3 of APR1400.

First of all, I think my presentation will proceed that these contents.

The section overview is briefly introduced and then each section is summarized as follows.

Chapter 3 consists of certain subsections from 3.1, Conformance with NRC General Design Criteria, through 3.13, Threaded Fasteners.

First of all, I'll address the key factor outlines of the DCD, Section 3.1 through 3.6 as a representative presenter.

If there is a question or comments, the answers will be provided by our responsible and expert staff sitting on my left and in this room.

The rest of this DCD Section 3 will be mentioned by the following presenters. Section 3.7 is Yongsun Lee, 3.8 is Hoonin Cho, 3.9 is Hongsun Park.

And, 3.10 through 3.12 is Bosung Choi. Lastly, 3.13 is presented by Hongsun Park.

For the NRC staff review, we only submitted to this DCD document and 11 TRs, Technical Evaluation Reports.

Now, I'm going to talk about the APR1400 conformance with NRC general design criteria in Appendix A to 10 CFR 50.

This section discussed the extent to which the design criteria for the plant structures, systems and components important to safety meet the NRC General Design Criteria specified in Appendix A to 10 CFR 50.

For each criterion, a summary was provided to show how the principle design features meet the criterion in the relevant DCD sections.

The Section 3.2 describes classification of SSCs according to seismic category, quality groups, nuclear safety classifications.

The Subsection 3.2.1 describes the classification of the SSCs in terms of seismic activity. It used the guideline provided in the Reg Guide 1.29 to meeting general design criteria, two, for identifying and classifying those SSCs.

These SSCs are classified as Seismic Category 1, 2 and 3.

Subsection 3.2.2 describes the various equipment classifications based on -- identified to meet GDC 1.

Quality Group A, B, C and D are classified by Reg Guide 1.26.

3.2.3 describes the safety classes. Fluid system components important to safety are classified in accordance with ANSI/ANS-51.1-1983 as Safety Class 1, 2, 3 and non-safety class.

MEMBER SKILLMAN: Before you proceed, back to 3.2.2, you explained quality classifications A, B, C and D. Would you care to make a comment about quality groups E and G?

You actually had six quality classification groups.

MR. YOON: Yes.

MEMBER SKILLMAN: You put A, B, C, D and then you also have E and G.

MR. YOON: Yes, we considered quality groups as poor, A, B, C, D, as I mentioned, but -- excuse me, I'm not the right person to answer or to provide expert steps.

MR. CHOI: My name is Bosung Choi. I am KEPCO, Inc.'s Licensing Engineer. I'll reply to your question.

Two other quality groups, E and G, is classified to indicate the governing design course for confidence. So, it is not covered by Reg Guide 1.26.

MEMBER SKILLMAN: Do you have the six quality classifications for your Korean plant?

MR. CHOI: Yes.

MEMBER SKILLMAN: Yes, so, you follow A, B, C, D, E and G for them as well?

MR. CHOI: Yes, right.

MEMBER SKILLMAN: And E and G are exception categories so that you can apply seismic qualification to components that aren't directly associated with A, B, C, and D? It appears that that's what you've attempted to do.

MR. CHOI: Yes.

MEMBER SKILLMAN: Will we have a chance to hear you speak about your emergency diesels and how they are addressed from a seismic perspective? I believe that is the category G.

MR. JEONG: This is Hyeok Jeong from KEPCO EDG.

The seismic category of EDG is seismic category 1. And we apply the quality group G because the quality standard applied to EDG is different from the quality standards defined in Reg Guide 1.26.

G, the quality group G is applied to the safety-related systems and components that the quality standards applied to different from the Reg Guide 1.26.

MEMBER SKILLMAN: Different from 1.26? I understand.

MR. JEONG: Thank you.

MEMBER SKILLMAN: We'll stop here. Okay, thank you.

MR. YOON: The Subsection 3.2.4 is the risk component classifications. This detail of 2.3.2-1 describes the seismic categories quality groups nuclear safety classifications, quality assurance classifications and quality standards.

In this slide, it talks about wind and tornado loading which is considered in the design of seismic category 1 and 2 structure.

The design wind loading on the surface of seismic category 1 and 2 SSCs subject to wind are determined in accordance with ASCE/SEI 7-05.

Fifty-year 30 second wind gust speed is determined to 145 miles per hour which corresponded to the wind speed at the 33 feet above the ground for exposure category C.

The APR1400 standard and site specific plant is designed to protect SSCs listed in the Appendix to Reg Guide 1.117 from tornados and hurricanes.

The applicable design parameters for tornados and hurricanes is as follows for design base tornados, maximum horizontal wind speed for 10 million years is 213 miles per hour as those are list one and categorized in Reg Guide 1.76.

For design basis hurricanes, maximum wind speed for 10 million years is 260 miles per hour. That is the demand from the wind speed control maps for hurricane prone regions of the parameters in 19 states presented in Reg Guide 1.221.

In this slide, I'll talk about the standard design basis flood level which is considered at the SSCs design.

The design basis flood level at reactor site will be determined in accordance with Reg Guide 1.59 and ANSI/ANS 2.8.

The design basis flood level of the APR1400 is, at the least, 1 feet below the plant grade.

Accordingly, all safety related SSCs located on the reactor site are protected from the external flood event.

The maximum ground water levels is 2 feet below the plant grade, all seismic category 1 structures are designed to withstand the static and dynamic forces due to the maximum ground water level.

The flood protection measures for seismic category 1 SSCs from external sources design in accordance with Reg Guide 1.1 or 2.

From this slide, I will speak about the flood protection from internal sources.

The safety related SSCs are designed to withstand the effect of flooding due to the natural phenomena or onsite equipment failures without the loss of the capability to perform their safety related functions.

The flood protection mechanisms from internal sources are designed in APR1400 in consideration of structure enclosure or barrier walls, a drainage system including Emergency Overflow Line, this is called the EOL, emergency sump, internal curbs or ramps and watertight doors.

Potential flooding sources include flood water due to the postulated pipe failures, inadvertent operation of fire protection systems and failure of non-seismic piping.

This slide is about the variation of internal flooding analysis for reactor containment building and auxiliary building.

For reactor containment buildings at the flooding sources flow to the elevation 100 feet of the reactor containment building and flow to the hold-up volume tank. And the containment building, water storage tanks through 24 inch spillways.

The limiting flood sources is water source from LOCA scenarios.

The flood height is determined to be two feet from the elevation 100 feet, considering the floodable area and LOCA volumes.

For auxiliary buildings, the flood sources flows to the bottom level at elevation 55 feet by drainage system and emergency overflow line.

The limiting flood source is the water

source of the IRWST.

The flood height is determined to be nine feet from the elevation 55 feet considering total water volumes of IRWST and floodable areas.

Each quadrant of the auxiliary building at elevation 55 is separated by division walls. This elevation is the lowest elevation in the auxiliary building.

And, the watertight doors are designed to prevent flooding source from spreading to adjacent quadrants.

MEMBER STETKAR: I'm surprised that you identified the IRWST as the limiting flooding source for the auxiliary building when your PRA identifies the fire protections systems as the most important flooding source in the auxiliary building.

MR. YOON: Yes, right. We have investigated the various internal flood sources in the containment building. There are many sources such as --

MEMBER STETKAR: I didn't say containment, I said auxiliary building.

MR. YOON: Yes.

MEMBER STETKAR: Okay.

MR. YOON: It's the same, sir.

There is failures flood source, as you mentioned, fire protection systems and feed waters and something like that.

But, the IRWST source is used for barriers function to feed water primary system and this is the limiting sources in auxiliary building.

MEMBER STETKAR: Okay, go on.

MR. YOON: Maybe I'm not articulating it but, the volume is --

MEMBER STETKAR: As I recall, I'm trying to read my notes here quickly, I didn't recall the IRWST being identified in the PRA as a flooding source for the auxiliary building.

And, I recalled, and in fact, my notes confirm that, that the fire protection system was the most important flooding source in the auxiliary building.

So, I'm curious why the presentation today now says that the IRWST is the most important source or the limiting source.

MR. YOON: As this case is most elevation auxiliary building --

MEMBER STETKAR: Sure, 55 meters --

MR. YOON: Yes, yes.

MEMBER STETKAR: Or 55 feet.

MR. YOON: Yes.

MEMBER STETKAR: Yes, well, I don't do well in plumbing and I -- the only think I know about plumbing is water flows downhill as does all of the fire protection system water flow downhill.

So, I'm just curious why we're hearing about the IRWST today when the PRA does not mention it and why we're not hearing about the fire protection system today when the PRA identifies the fire protection system as the most important flooding source in the auxiliary building.

MR. YOON: In my --

MEMBER STETKAR: So, I'm just curious about that. I don't think we'll have an answer today.

MR. SISK: This is Rob Sisk, Westinghouse.

Appreciate the question and let us take that back and look into that.

MEMBER STETKAR: Sure, thank you.

MR. YOON: Let me go to the next slide, please?

From this slide, I'll speak about missile protection regarding to this Section 3.5.

The analysis has been performed to meet the regulatory requirement 10 CFR 50 Appendix A GDC 2 and 4.

Missile protection is provided for SSCs important to safety so that the potential missile cannot cause the release of significant radioactivity or do not prevent the safe shutdown of the reactor.

Missile protections are accomplished as follows. First, to minimize the missile generation source by equipment design.

And, second, to orient of physically separate the potential missile source from safety related equipment and components.

Third, to contain the potential missiles through protective shields or barriers.

Fourth, hardening the safety-related equipment and components to withstand the missile impact.

Internally generated missiles could be categorized as two types of components which are rotating and pressurized components.

The probability of missile generation is used for validating the internally generated missiles.

If the probability of missile generation P1 is maintained less than 10-7 per years, missile is not considered statistically significant.

Potential internal missile sources inside containment are listed in Table 3.5-1.

To protect the SSCs from internal missiles, structure inside containment, secondary shield wall, refueling pool wall, beams and slabs are served as missile shields.

It's identified in Subsection 3.5.1.2.

Concerning the turbine missile, the analysis has been performed based on Reg Guide 1.110 and the related SRPs.

The turbine generator is placed with favorable orientation so that all essential SSCs are excluded from the low-trajectory turbine missile strike zone as shown in this figure 3.5-1.

Essential SSCs to be protected from external generated missiles and turbine missiles are listed in Table 3.5-4.

For missiles generated by the tornados and extreme winds, safety-related SSCs of the APR1400 are protected against the impact generated by tornado or hurricane missiles.

The protection provides reasonable assurance of performance with related 10 CFR requirements.

The five automobiles and solid steel spears are considered as a typical external missiles.

MEMBER SKILLMAN: Let me ask you a

question about that. As I read the safety evaluation report that is associated with your Chapter 3 Revision 1, there is a statement that there's a COL item that will address that automobile missiles cannot be generated within a half-mile radius of safety-related SSCs that would lead to an impact higher than 10 meters.

What is the basis of that?

MR. YOON: Yes, I think that answer is we provide specialist in the head.

MEMBER SKILLMAN: That would be a very interesting answer. We've got tornados in this country and high winds in this country that can certainly move a pickup truck or an automobile 2600 feet, can deposit it on a roof of the building.

So, I'm wondering about the accuracy of that assumption.

That would also presume that there's an exclusion radius around safety-related equipment that precludes vehicles being closer than a half-mile.

MR. YOON: Yes.

MEMBER SKILLMAN: And, most of the plants that I've seen have automobiles, pickup trucks, delivery trucks closer than a half-mile.

MR. NOLAN: This is Ryan Nolan from the

staff.

That's actually an item that's in the guidance, Reg Guide 1.76 and Reg Guide 1.221. So, that's where the --

MEMBER SKILLMAN: Half-mile came?

MR. NOLAN: -- the 10 meters came from. And then, the COL item is maybe I'll call it operating experience for lack of a better term. We had some COL sites where they had elevated parking lots.

And so, we wanted to ensure that an automobile missile would not invalidate that 10 meter assumption. So, that's sort of the history of where that comes from.

MEMBER SKILLMAN: Good, thank you. Thank you.

MEMBER STETKAR: I need to, before we continue, correct something that I said on the record.

I don't know what the most important flooding contribution is in the updated version of the PRA. But, as I scan through Chapter 19 of Rev 1 of the DCD, I noticed that you do not -- that you now do take account for breaks that drain the IRWST into the auxiliary building.

So, that's -- it's included in the scope of the model to now, it's just I haven't seen them.

So, forget the last question, it's at least in there, it's somewhere.

Thank you.

MEMBER REMPE: So, I had a question, too, on this section. I found the wording in the design certification document as well as in the staff's SC about bit vague about the multi-unit issue.

And, is it planned somewhere that -- I mean, it would have helped if the staff had clearly -- and I was going to bring it up to them, instead, this is for a single unit application. But, instead, it's bouncing that off again about the considerations of the -- in the -- about multi-unit sites.

And, I was just kind of curious if KHNP plans to clarify any of the discussion in updated versions of this chapter?

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

For this application is only one unit that persist. But, in terms of multi-unit sites for the missile protection or turbine generators missile protection assessment is performed. We've set it as a COL action item for COL action item 3.5-2.

So, COL applicant is perform assessment of the -- those assessment when, as a multi-unit established in the same site construction.

MEMBER REMPE: Yes, I guess I'd like to see the wording clarified because it -- I think it can be misinterpreted and I would like, especially with the staff, to just clearly say, hey, again, this is a single unit application and if there are other existing units prior or after for other types of units, that's -- you need to consider this.

But, I think I've made the point hopefully.

MR. OH: Correct.

MR. SISK: This is Rob Sisk, Westinghouse.

I will point out that it does say in the DCD, I'm looking at language here, specifically that SSCs that perform safety functions are not shared between two units because the APR1400 is a single unit plant.

MEMBER REMPE: In what chapter?

MR. SISK: Chapter 3.1.5, just as an example.

MEMBER REMPE: Okay. But, later in like Section 3.5.1, it gets a little more vague. But, I didn't see that it said that earlier. So, that's good, at least in your section it clearly says it.

MR. SISK: Well, I just wanted -- it is in various sections --

MEMBER REMPE: Yes?

MR. SISK: -- various chapters. And, where there are potentials for multi-unit, whether new construction or existing construction --

MEMBER REMPE: Right.

MR. SISK: -- there is a COL item for looking at the site and evaluating --

MEMBER REMPE: Okay.

MR. SISK: -- proximity to other sites. So, it is in there, I'm not sure if it's everywhere that you would like in terms of the chapter, but we do try to be clear that it's a single unit application.

MEMBER REMPE: Okay.

MEMBER STETKAR: We've actually seen COL applicants that had to address that when they've installed either a single unit where turbine -- I'm talking about main turbine missiles now, could affect one or more of the existing buildings and --

MEMBER REMPE: I do think --

MEMBER STETKAR: -- even a dual unit installation where missiles from one of the units can affect part of, you know, they're within the -- words fail me.

MEMBER REMPE: I get what you --

MEMBER STETKAR: Within the area.

MEMBER REMPE: I get that you need to think about it for existing or new construction, but it just sometimes it --

MEMBER STETKAR: And people have -- my point is, people have. I mean, we've seen COL applicants who have actually had to deal with that.

MEMBER REMPE: Yes, I --

MR. YOON: And, let's get started, okay?

On this page, also we considered the site proximity missiles, the COL applicant is to evaluate the potential for site proximity explosion and missiles.

Due to the train explosion, shipboard explosion, in the steel facilities, pipeline exclusion or military facilities.

The COL applicant is to provide justification for the site specific aircraft hazard and an aircraft hazard analysis in accordance with the requirement in Reg Guide 1.206.

The aircraft impact analysis is address in Section 19.5.

For the structures, systems, components to be protected from externally generated missiles such as used to protect safety-related SSCs met the requirement Reg Guide 1.13, 1.27, 1.115 and 1.117.

The essential SSCs protected against a missile impact are listed in this Table 3.5-4.

This slide is for as assist to be protected from externally generated missiles and barriers design procedures.

Missile barriers are designed with sufficient strengths and thickness to prevent local damage, including perforation, spalling and spreading and overall damages.

Local damages prediction for concrete structures includes the estimation on the depths of a missile penetration and on assessment of whether a secondary missile could be generated by spalling.

The steel barriers are not used in the APR1400 design to protect safety-related equipment or systems from missile impact.

For the variation of overall expanse, reinforced concrete barriers and impact and impulse load, nonlinear and elastoplastic response of structures is used.

This slide shows the dynamic effect associated with the postulated rupture of piping.

Plant design break analysis has been performed in accordance with the following regulatory requirements such as 10 CFR, SRP and ANSI/ANS.

The high and moderate energy fluid systems are identified and summarized in this Table 3.6-1.

The protection design such as separation, physical barriers or pipe heat restraints are designed to protect essential SSCs from the effect of postulated pipe break.

The pipe break locations are determined in accordance with BPT 3-4, Part B.

Break locations are considered at terminal ends and at the intermediate location based on stress analysis results.

The criteria used to define location and configurations of pipe ruptures are provided in Subsections 3.6.2.1.

The analytical method to define forcing functions are described in this Subsection 3.6.2.3.

The Main Steve Valve House is the only area that met the break exclusion criteria described in BTP 3-4, B.A(ii) for ASME Class 2 piping.

The design of break exclusion is applied for fluid system piping and containment penetration areas, between containment and auxiliary building anchor walls beyond the isolation valves.

The forcing function for pipe thrust and jet loads are based on ANSI/ANS 58.2, Appendix B procedure addressing the methodologies.

The dynamic and environmental impact due to the HELB and MELB are summarized in the pipe rupture analysis report.

The dynamic impact is analyzed with respect of pipe whip, jet impingement and sub-compartment pressurizations.

The environmental impact considered with regard to flooding and environmental qualifications.

The potential non-conservative of jet impingement models in ANSI/ANS 58.2 will be addressed in some open items.

With respect to blast waves, jet plume expansion and zone of influence and distribution of pressure within the jet plume and jet dynamic loading including potential feedback amplifications and resonance effects.

The current technical report and related RAI responses will be revised to address the above issues.

In this slide, I'm going to talk about structural design of pipe whip restraints.

PWRs provide to protect the safety-related components against impact of pipe whipping during the postulated pipe break. PWR is pipe whip restraint.

The design of pipe whip restraint is governed not only by the pipe break blowdown thrust, functional requirement, deformation limitations but also by the property of the whipping pipe and the capacity of support structures.

The strain of energy-absorbing members is limited to 50 percent of ASTM specified minimum elongations.

To allow stress for non-energy-absorbing members, structural attachment and support steel structures are specified in AISC N690.

The consider of leak before break is applied to APR1400, the LBB analysis is used to eliminate the dynamic impact of pipe break.

This section describes how the piping system with the LBB criteria and demonstrate that the probability of pipe ruptures is extremely low.

The design of the LBB is incorporated into the reactor coolant loop piping, a surge line, direct vessel injection line and shutdown cooling line.

The method of piping evaluation diagrams allows for the evaluation of the piping systems.

The LBB evaluation is consistent with the requirement of SRP and NUREG report.

MEMBER RICCARDELLA: Yes, approximately, what percentage of the Class 1 piping was addressed via leak before break versus dynamic analysis?

MR. YOON: We -- the concept of LBB is applied to various systems and specifically into the primary system. And then --

MEMBER RICCARDELLA: But, I mean, is it half of the systems usually leak before break? Ninety percent of them? You know, that's what I'm trying to get a feel for.

MR. YOON: Excuse me, let me talk about coolant and guides.

MR. SISK: This is Rob Sisk, Westinghouse. We'll take that offline and get a discussion. We don't have a precise or a good estimate at this point.

MEMBER RICCARDELLA: Yes, I don't need it precise, I just wanted a feel.

MR. SISK: Yes, I think we need to get the right individuals.

MEMBER RICCARDELLA: Thank you.

MR. Y. KIM: Yes, Yunho Kim.

Actually, we cannot say the percent of piping for LBB application, just to be saying -- that we can say that main reactor coolant system and the surge line that is reapplied to LBB. But, I cannot say how percent of pipe, I don't have the data.

MR. YOON: This is the end of Section 3.6. If there is no more questions or comments, let me transfer the speaker to Yongsun Lee. He will take up Section 3.7.

MR. Y. LEE: Good morning, ladies and gentlemen, I am Yongsun Lee, Senior Engineer of KEPCO EDC Seismic Team.

I am going to present Chapter 3.7, Seismic Design including Seismic Design Parameters, Seismic System Analysis, Seismic Subsystem Analysis and Seismic Instrumentation.

For Section 3.7.1, Seismic Design Parameters, this slide presents certified seismic design response spectra.

The peak ground acceleration of the CSDRS is set to 0.3g for the APR1400 design for the boost parameter and vertical directions.

The horizontal and vertical CSDRS for the APR1400 are based on the Regulatory Guide 1.60 response spectra.

In each of the high frequency range from 9 hertz to 50 hertz.

Below figures show the horizontal and vertical CSDRS with 2, 3, 4, 5, 7 and 10 calculations.

This slide presents time histories compatible with CSDRS. You notice with Option 1, which 1 standard plan 3.7.1.

The three design histories and time histories which envelope the CSDRS are generated to be applied in both surge injection lines and system based analysis of the APR1400 Seismic Category 1 structures.

The initial motions that were used to create the design time analysis was actual recorded 0.3g plus K time histories.

The cross-section coefficients among the three design CSDRS time histories are less than 0.16 as specified in standard plan 3.7.1.

The figures show the east, west, north, south and vertical components of the design affiliated time histories from the top.

This slide presents the high frequency response spectra.

The groundwater and response spectra for some Central and Eastern United States rock sites show higher amplitude at the high frequence standard CSDRS.

The ground response -- the ground motion response spectra for such a site, called HRHF seismic motion, the peak ground affiliation HRH -- the peak ground acceleration of the HRHF response spectra prescribed as 0.46g for the evaluation of the APR1400 standard plant design for both the horizontal and vertical directions.

The HRHF horizontal and vertical response spectra are shown in the figures. This HRHF response spectra, you see the CSDRS for the frequencies higher than approximately 10 hertz.

This slide presents the HRHF time histories. The three HRHF time histories which envelope the HRHF response spectra also generated in accordance with Option 1, approach 1 in standard review plan 3.7.1.

The initial motion the vertical pipe creates the time histories compatible with HRHF response spectra are actually seed recorded Nahani plus K time histories.

The cross-correlation coefficients are among these HRHF associated time histories are also less than 0.16.

The figures show the east, west, north, south and vertical components of the HRHF time histories from the top.

This slide presents the critical damping measures. Basically, damping measures is used for various nuclear safety-related structures, systems, components, based on the Regulatory Guide 1.61.

The damping barriers of soil using the soil-structure interaction analysis obtained from the generally models reduction and damping covers recommended by EPRI report.

The table shows the various structures damping related using the seismic soil structure interaction analysis and peak space analysis.

This slide presents the supporting media for the Seismic Category 1 structures.

Seismic Category 1 structures are founded directly on rock or component soil.

For the design of Seismic Category 1 structures, eight soil profiles and one fixed-based condition are established with various velocities compared with soil depth to cover a wide range of seismic conditions.

Figures show seismic velocities of several soil profiles using the seismic analysis, for example.

The soil velocities, blue lines in the figure and compression velocities, red dot lines in the figure, varies from approximately 1000 feet per sec to 9200 feet per sec and from 4800 feet per sec to 18,000 feet per sec respectively in the soil profiles.

For Section 3.7.2, Seismic System

Analysis, this slide presents the APR1400 seismic analysis methods and models.

Complex frequencies response analysis method is used for the three dimensional soil structure interaction analysis and fixed-based analysis to obtain seismic response of the Seismic Category 1 structures.

A fixed-based analysis also performed with this complex frequency response analysis method by assigning a very high seismic wave analysis to the foundation medium.

The APR1400 safety-related structures such as reactor containment building and auxiliary building are modeled as three dimensional finite element models.

This slide shows the seismic analysis models for the auxiliary building and emergency diesel generator building and the diesel fuel storage tank room.

For the reactor containment building seismic model, please refer to DCD R2, figure 3.7-24 and 25.

This slide presents the in-structure response spectra generation.

The complex frequency response method with

ACS SASSI software is used to generate ISRS at wall and floor locations in the finite element models.

For the impact of potential concrete cracking, both uncracked and cracked concrete stiffness cases are considered separately in the seismic analysis models.

So, the finite design basis ISRS envelopes the ISRS generated from 9 soil cases included fixed-base analysis condition for both uncracked and cracked concrete stiffness cases.

This slide presents the interaction non-seismic Category 1 structures with the Seismic Category 1 structures.

The turbine generator building and compound building are located on the west side and the south side of the auxiliary building with a 3-feet gap on each side.

The Seismic Category 2 turbine generator building and the compound building are to be analyzed and designed to prevent the failure under the SSE condition in the COL stage.

To evaluate the structure soil structure interaction impact on the nuclear island due to the presence of adjacent non-seismic Category 1 structures such as turbine and compound buildings, the structure soil structure interaction analysis using the coupled model for the entire power block is performed.

The interaction impacts of this non-seismic Category 1 structures on the nuclear island are negligible. However, the nuclear island impact or seismic response of the emergency diesel generator building source, the ISRS of the emergency diesel generator building are increased to accommodate those interaction impacts.

This slides shows the structure soil structure interaction analysis model for the entire power block with surface-supported foundation condition.

And, this slide shows the structure soil structure interaction analysis model for the nuclear island and emergency diesel generator building with excavated soil volumes.

This model is to consider embedded foundation conditions.

MEMBER SKILLMAN: I would like to ask this question, you just mentioned that the compound building and the other buildings will be the responsibility of the COL applicant.

MR. Y. LEE: Yes, sure. The design of the turbine generator building and the compound building is not included in the APR1400 standard design. So, those buildings should be analyzed and designed in the -- by the COL applicant.

MEMBER SKILLMAN: The reason I asked the question is, it seems that there are systems and components in the compound building and the essential service water building and the alternating current building that are critical components to the function of the nuclear island.

And, I'm excluding the turbine building because the turbine building could be a kilometer away if you wanted to transport the steam.

But, the compound building and several of these other buildings are the structures that protect components that are used in the nuclear island.

So, my question is a broad question, why aren't those particular buildings part of your design application? Right now, they are left to the COL applicant.

MR. Y. LEE: My understanding is that major important systems are designed in this APR1400 standard plant design. Those are included in the center design.

But --

MEMBER STETKAR: That's not quite true

when you look at the -- I always forget the name of it, you guys call it essential cooling water -- the service water.

The structures for the service water, the component cooling water heat exchanger building and the piping tunnels between the nuclear island and those other buildings are left to the design of the COL applicant.

MR. Y. LEE: Yes, those are also classified as Seismic Category 1 structures. But, those structures are also -- do not included in the standard design scope.

MEMBER SKILLMAN: I understand that they are excluded. I'm asking why. And, my point is, as John Stetkar points out, the UHS building, the component cooling water tunnels, the component cooling water heat exchangers are all somewhere else and not in the nuclear island.

But, those devices, those systems and structures, are critical to the successful functioning of the nuclear island and the structures in which they reside are, therefore, highly important.

So, my question is a curiosity question more than a challenge. But, I'm wondering why those key systems and their buildings are not part of your standard design?

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

For example, for the ESW building, for a COL applicant can locate the ESW building, depending on a site specific condition. So, the -- as per the standard design applicant, we cannot locate the specific condition. That's the reason and for the counter for the CCW.

If the COL applicant is positioned for the north way or south way, and depending on the tunnel where it can be determined.

So, that's the one reason for the standard design applicant not decided for the tunnel, the building or something.

And also, for ESW for essential service water is very specifically depending on the site specific condition.

MEMBER SKILLMAN: I understand that, Andy. But, it seems to me that those buildings that I mentioned are as critical to the successful operation of the nuclear island as your emergency diesel generator building.

MR. OH: Yes, I --

MEMBER SKILLMAN: I put them in the same

category. And so, it seems to me, that because of the design specific requirements for those systems, structures and components, KHNP would have insisted that those key buildings and their components would have been governed by your standard design.

MR. OH: Right, right, sir.

So, for system design is in scope for the APR1400. Essential service water and component cooling water and also even in a compound building for waste assistance.

System design is in scope for this standard design. However, for the structural design for the seismic Category 1 or something, it's just for the COL applicant. You have to construct the building seismic will be classified for the Category 1. That's per our request to standard design and requested for COL applicant.

So, that's the current status for the APR1400 designs.

MEMBER SKILLMAN: Thank you. I understand your answer, thank you.

MR. Y. LEE: Thanks, Dr. Oh.

This slide presents the incoherent SSI analysis to evaluate the impact of the HRHF seismic input motion.

Based on the 2011 EPRI report, the APR1400 HRHF response spectra determined that the 0.8-fractile, five percent depth for the entire containment site and below ground motion response spectra for the Central and Eastern United States hazard size.

The soil structure interaction analysis is performed with incoherent input motion and coherency functions developed by Norm Abrahamson using input code and the incoherent ground motion vector input to ACS SASSI software.

So, the seismic capacity of the structures and systems and components of the nuclear island, including reactor containment building and OCR building and the emergency diesel generator building are examined.

For the Section 3.7.3, Seismic Subsystem Analysis, this slide presents the seismic subsystem analysis method.

The seismic analysis of Seismic Category 1 subsystem is performed using either the response spectra analysis or time history analysis or equivalent static analysis method.

Since a lot of the content is Section 3.7.3 are similar to those in Section 3.7.2, the content in Section 3.7.3 are not described in this presentation model.

For Section 3.7.4, Seismic Instrumentation, this slide presents seismic monitoring system of the APR1400 standard plant.

The seismic monitoring system is designed in accordance with Regulatory Guide 1.12.

The time history accelerographs installed at appropriate locations which are described in Regulatory Guide 1.12.

Provide time history data on the seismic risk parts of the pre-fueled contained structure and other Seismic Category 1 structures.

MEMBER SKILLMAN: What are you going to use this system for?

MR. Y. LEE: For the seismic monitoring system?

MEMBER SKILLMAN: Yes. What are you going to use it for?

MR. Y. LEE: That --

MEMBER SKILLMAN: Okay, let me say a little more. You are using an OBE that is one-third your SSE.

MR. Y. LEE: Yes.

MEMBER SKILLMAN: Is it your intention

that if your seismic monitoring equipment shows that the ground motion is one-third of SSE or less, therefore, OBE, that you simply do an inspection and keep on operating? Is that your -- is that the design intent of your seismic monitoring system?

MR. Y. LEE: Yes, basically, the OBE data is set to the seismic monitoring system to investigate -- I'm sorry -- basically, the core nuclear part plant has the seismic monitoring system based on the OBE, that slide.

And, the OBE is now set to the one-third SSE.

MEMBER SKILLMAN: Okay. So, let's imagine that we are in the control room and there is ground motion and the ground motion is just a little bit less than the OBE. What do we do?

MR. Y. LEE: The ground motion is not exceed the OBE, am I right?

MEMBER SKILLMAN: You're right.

MR. Y. LEE: In that case --

MEMBER SKILLMAN: You're right, okay. Just keep on going?

MR. Y. LEE: I will pose that it's like that, but the details should be prepare -- should be described in the operation manual.

MEMBER SKILLMAN: Yes, okay. I was just trying to understand what you intend to do with your one-third SSE which is your OBE, which is appropriate under Appendix S of 10 CFR 50.

So, I understand what you're communicating. I was just wondering, what is the design intent? Is your seismic monitoring system intended to demonstrate overall plant experience with the ground motion? Or, is it intended to be an operator's aid to enable continued operation after minor ground motion?

MR. Y. KIM: Yes, this is Yunho Kim from KHNP.

Actually, we had also in Korea listen to the variety of the model. Little --

MEMBER STETKAR: Andy, speak up a little bit so -- or, I'm sorry --

MR. Y. KIM: Yes, so actually, we had experience that we also can listen to it in the Kenji site as compared to our Shinkori site.

And so, the main purpose of the seismic monitor is just to input seismic is exceed one-third of SSE which is OBE, then we have a procedure to -- for the operator to trip the reactor.

And then, after trip, there is a procedure to look around to see the condition.

MEMBER SKILLMAN: To inspect?

MR. Y. KIM: Yes, yes. So, what is the impact on also to major safety components, we want to know. So, that is the, I think, main purpose of the seismic monitoring system to aid for the operator to trip the system.

MR. OH: Yes, this is Andy Oh, Washington Office.

One more thing is for the monitor is a recorder function. So, from that, the motions, then after that, we can just evaluate how much impact is to the plant.

And so, it's like the Member Skillman said, that case port. It's very close to the OBE, but it's not exceeding it. But, in that case, we also consider, is there some of the, you know, additional action we have to take for inspection for the plant or not.

That is also can be the function for this seismic monitoring system.

MEMBER SKILLMAN: Thank you, thank you.

MR. Y. LEE: There is one open item related -- important to the analysis of the HRHF seismic motions in Chapter 3.7.

We are working on this and try to resolve it.

This is the end of my presentation for Chapter 3.7. Thank you for your attention.

Do you have any further questions or comments?

Then, the next chapter, 3.8, will be presented by Mr. Hoonin Cho.

Thank you.

MR. SISK: Maybe just switch out.

MR. CHO: Sorry, good morning. I'm Hoonin Cho and I'm in charge of ICS Structural Design in KEPCO E&C.

Now, I'm going to talk about 3.8 design of Category 1 structures.

First of all, let me tell you about concrete containment structure description.

The containment is a pre-stressed concrete structure composed of a right center cylinder with a hemispherical dome and is bounded on safety-related common basement.

The structures are lined on the inside of a quarter-inch thickness steel plate that acts as a retard membrane.

The internal structures are basically

independent of the containment except at the supporting foundation basement.

The containment shares a common basement with the auxiliary building. The auxiliary building wraps around the containment with a seismic isolation gap of six inches.

There are a equipment hatch, two personnel airlocks, IRWST, tendon gallery and polar crane.

This slide shows containment dimensions. Inside diameter of containment, inside height of containment from the top of base slab to the ceiling of dome apex, the thickness of containment wall and the dome thickness.

The 3D Finite Element Model with ANSYS was developed to perform the structural analysis of the containment structure.

MEMBER SKILLMAN: May I ask this question before you proceed?

The tendons --

MR. CHO: Yes?

MEMBER SKILLMAN: -- that will be used are approximately 100 meters long, as I calculate.

MR. CHO: Yes.

MEMBER SKILLMAN: They are continuous?

MR. CHO: Yes.

MEMBER SKILLMAN: How are the temperature changes associated with those tendons accounted for in the analysis of the containment? That member is so long --

MR. CHO: Yes.

MEMBER SKILLMAN: -- and there are so many of them, the changes from summer to winter, and particular summer heating on the concrete will cause the tendons to expand and contract ever so slightly unless the spring constant is such that the tension is relatively constant than the compression of the building is changing according to those temperature changes.

And, I have the same question relative to accident maximum temperature when those tendons would, therefore, be relaxed.

So, my question is, these tendons are so long, how is the temperature change with the tendons accounted for?

MR. CHO: Yes, actually, in the analysis of the structure, the temperature change is considered in the concrete members.

As you know, the temperature is various through the thickness. But, actually, the temperature change in tendon is not considered because we think it is in the middle of the wall of the containment and it may be negligible inside the containment concrete wall.

MEMBER SKILLMAN: How certain are you that that temperature change is negligible?

MR. CHO: In tendon?

MEMBER SKILLMAN: Yes. Let's say a 3 degrees C, 2 degrees C temperature change in a 100 meter long tendon --

MR. CHO: Yes.

MEMBER SKILLMAN: -- could result in a more than inconsequential change in tensile force.

MR. CHO: I'm sorry.

MR. SISK: This is Rob Sisk, Westinghouse.

I think we understand the question being asked relative to the small temperature change and the confidence level relative to the length of the tendon.

MEMBER SKILLMAN: Yes.

MR. SISK: We're going to take that aside and look into that in a little bit more detail.

MEMBER SKILLMAN: Fair enough. And, maybe in a topical report or technical report --

MR. SISK: We got you.

MEMBER SKILLMAN: -- that I think that the assumption that the change will be -- the temperature change will be negligible is one that may have some significant uncertainty associated with it.

If we were talking tendons that were 30 meters long, 40 meters long, I wouldn't ask the question. But, these tendons are very long tendons. Perhaps there's experience, it's just experience I've never had, but I'm curious.

MR. SISK: And we need to check into that and make sure we have it correctly understood. So, let --

MEMBER SKILLMAN: Okay.

MR. SISK: -- we'll take that as a --

MEMBER SKILLMAN: Let's make an open -- an action item, please?

MR. CHO: Okay.

MEMBER SKILLMAN: Thank you.

MR. CHO: Let me go on?

MEMBER SKILLMAN: Please.

MR. CHO: For combustible gas control inside containment, structural integrity of the containment structure was evaluated in accordance with Reg Guide 1.216.

Acceptance criteria is provides in ASME Code CC-3720.

Maximum temperature of 350 degrees

Fahrenheit was used to evaluate according to NUREG.

The 3D Finite Element Model were developed with ABAQUS software to confirm the structural integrity of the containment structure and the combustible gas load condition.

3D Finite Element Model consists of the three types element for four materials.

The maximum pressure is 109 psig.

Analysis research shows the maximum liner plate strain for tension and it does not reach the allowable limit strain values based on ASME CC-3720.

The assessment of the pressure capacity of the containment were performed based on the Reg Guide 1.216. At this pressure condition, the structural integrity is retained and a failure leading to a significant release of fission product does not occur.

Analysis research shows that the UPC pressure is 158 psig.

The research of the design of the pressure reinforcement are summarized in DCD.

For pressure reinforcement, it is confirmed that the maximum stresses provided reinforcement do not exceed the allowable stresses for both the service and factor load conditions.

Let me talk about the steel part of the

containment.

The COL applicant is to provide the detailed design results and the variation of the ultimate pressure capacity of penetrations including the equipment hatch, personnel airlocks, electrical and piping penetration.

Let me talk about structural descriptions of the internal structures.

The internal structure is a group of reinforced concrete structures that enclose the reactor vessel and primary system.

The internal structure is located in the reactor containment building, consists of the follow major components, PSW, SSW, IRWST, operating and intermediate floors and refueling pool.

The three dimensional finite element model with ANSYS software was developed to perform the structural analysis of the internal structure.

The results of the design are summarized in DCD.

The concrete section strengths determined from the criteria in ACI 349 are sufficient to resist the design basis load.

This slide shows the structural description of an auxiliary building. AB is reinforced concrete structures which is composed of a rectangular walls, floors, slabs, columns and beams.

AB surrounds RCB with a seismic gap of six inches and share a common basement with RCB.

AB structure provides a protection against both external and internal hazards.

AB is separated from other buildings by the isolation gap of three feet.

The auxiliary building is rectangular with the maximum dimensions of 348 feet by 353 feet.

AB structural analysis method is for static analysis. The analysis was performed to find all the member forces of shear and section shear forces of slabs in AB.

The result on the design for AB are summarized in DCD.

The concrete section strengths determined from the criteria in ACI 349 are sufficient to resist the design basis load.

This slide shows the structural description of EDG building and diesel fuel oil tank building.

EDG building block consists of two independent buildings. EDG is separated from other buildings by the isolation gap of three feet.

EDGB houses two additional generators and DFOT building houses the DFOTs thereof.

The results under design for EDG building are summarized in DCD.

The concrete section strengths determined from the criteria in ACI 349 are sufficient to resist the design basis load.

This slide shows structural description of NI common basement. NI common basement consists of two areas, one central circular shaped area which supports RCB and the other rectangular shaped area which supports AB.

Disk shaped reinforced concrete structure in RCB area has various thickness.

For the applied load on the NI basement analysis, the equivalent static acceleration method is used to consider the seismic load.

Stability check for overturning, sliding and floatation of NI common basement was performed in accordance with Section II of SRP 3.8.5 and the results are as follows.

All the results meet the SRP requirement as shown in this table.

The results on the design for RCB basement and AB basement are summarized in DCD.

The concrete section strengths determined from the criteria in ASME Section III Division 2 Subsection 3C for ASME basement and ACI 349 for AB basement are sufficient to resist the design basis load.

This slide is the end of Section 3.8.

There are seven open items in Section 3.8 and it is summarized in the open items summary slides.

For next slide, Section 3.9, will be presented.

I appreciate your attention.

MR. P. HONGSUN: Good morning, Chairman and ACRS Subcommittee. It is an honor to be here today.

I am Park Hongsun from KEPCO E&C.

Let's start with Section 3.9. This section provides the method of design dynamic testing and analysis for Class 1, 2 and 3 components and supports, including core support structures.

This slide shows the special topics for mechanical components. This subsection provides the transient using the design and quality analysis of ASME Code Class 1 components and reactor internals.

The design transients gives the information such as fluid system pressure, temperature, flow rate and frequency.

However, it does not cover the seismic loading and other mechanical loading.

In this section, service level and test conditions are addressed.

For APR1400, the 60-year design life and the effect of environmentally assisted fatigue is considered.

The frequencies of events traditionally categorized as a Service Level C condition are conservatively classified as a Service Level B condition.

A number of computer programs are used in the stress and structural analysis. The programs are verified and validated in accordance with the design control methods.

Those are consist with the quality assurance program described in Chapter 17.

And, for APR1400, experimental stress analysis is not used.

In order to consider the evaluation of the faulted conditions, the major components of the RCS are designed to withstand the pipe break loads.

The system or subsystem analysis is performed on the elastic basis.

Section 3.9.2 is for the dynamic testing and analysis of systems, components and equipment.

Reactor coolant system structure analysis --

MEMBER SKILLMAN: Before you proceed, may I please ask some questions on 3.9.1?

MR. P. HONGSUN: Okay.

MEMBER SKILLMAN: I would like to direct your attention or KHNP's attention to your DCD. It is your page 3.9-4, it is Section 3.9.1.1.1 and this is Revision 1 of your design control document.

And, you're explaining the Service Level A conditions. And, you introduce the topic that the Service Level A condition consists of 14 events identified in Table 3.9-1.

Now, 3.9-1 is your main table for your stress analysis for the reactor coolant system. And, in that table on 3.9.1, your Service Level A conditions show nine events, not 14 events.

And so, either the table has an error or the text in Chapter 3.9.1.1.1 is in error. And, I'm referring to Revision 1 of the DCD in both instances.

MR. P. HONGSUN: Okay, I don't have DCD now, so I will check the table and the subsentences.

MEMBER SKILLMAN: And, I have a number of questions that I believe are at a level of detail that is too fine for this meeting, but I'm -- my background is in SSS design and I went through this table very carefully. And, I do have some questions about the KHNP assignment of transients for Service Levels A, B, C and D.

And so, we might be wise to have an offline discussion, or if you'd like me to, I can --

MEMBER STETKAR: We can't do that.

MEMBER SKILLMAN: Okay. Let me ask just several questions. Okay?

MR. P. HONGSUN: Okay.

MEMBER SKILLMAN: You have in your Service Level A condition, normal events 1A and 1B, steady state operation with normal parameters and you identify a million five cycles for increases and decreases and those are changes between 5 and a 100 percent.

A portion of these are repeated under Service Level B and I don't understand how the transient can be in both categories, Service Level A and Service Level B.

Because, if I go to Service Level B, I get a very similar set of conditions. These are decreases in main feedwater temperature and increases in flow rate which are identical in terms of what drive these million five cycles for power changes.

So, it appears as though there might be double counting for the transients. So, would it be to your benefit to not double count, because, as you count double, you're increasing your eating into your U, your cumulative utilization factor.

MR. SISK: This is Rob Sisk.

Thank you for that. We will look into both the Service Level A, B, C, D and check to see if there are any corrections or clarifications that need to be addressed.

MEMBER SKILLMAN: I had questions on starting and stopping of the reactor coolant pumps at hot shutdown.

And, where I was really pointing my question is relative to which loop is started and when you can drive flow backwards into the downcomer back into the steam generator.

And, it isn't clear to me how the reverse flow temperature changes are accounted for.

If you're in Mode 3, if you have some heat in the system, perhaps you are identical in T hot and T cold. But, once you begin to move T hot above T cold and go to a pump start, you generate a rather rapid transient.

And, it would seem to me that those need to be accounted for somehow. And, those transients are quite rapid, the reactor coolant pumps are about 110,000, 120,000 gallons a minute each.

And, depending on how you move those pumps, you're going to move that water very rapidly.

MR. P. HONGSUN: Thank you for your comments, we will check your comments.

MEMBER SKILLMAN: Let me -- I have one or two more.

MR. P. HONGSUN: Okay.

MEMBER SKILLMAN: You've included one steam generator tube rupture in 60 years. One steam generator tube rupture in 60 years.

It would be fine if that is your actual experience and there is not a whole lot of experience with steam generator tube ruptures.

But, it would seem to me to be prudent to consider more than one, even though the transient itself is not particularly severe. It turns out that the radiological conditions are the greater issue.

But, for as many tubes as you have, for 60 operating years, that is a long, long time.

CHAIRMAN BALLINGER: I might comment that, if they were using Alloy-600, that would be wrong. But, since they're using Alloy-690, they haven't had, to my knowledge, a tube rupture with Alloy-690.

MEMBER STETKAR: Also, my comment that the operating experience worldwide is less than one tube rupture every 60 years, much less.

CHAIRMAN BALLINGER: That does not mean that they won't have one in 60 years.

MEMBER STETKAR: No, it doesn't.

CHAIRMAN BALLINGER: It means --

MEMBER STETKAR: It's just that the average frequency is much lower than that.

CHAIRMAN BALLINGER: It's a much better situation.

MEMBER SKILLMAN: My final question is on your Table 3.9.2, I can pick it up in the next section when you get into 3.9.2, so I'll wait for you to move into 3.9.2.

MR. SISK: Well, we do thank you for the comments. And, we have captured those for consideration as we move forward.

So, please, go ahead.

MR. P. HONGSUN: Okay. Let's move on to Section 3.9.2, the reactor coolant system structural analysis is performed to generate the design data such as processes and movement, motions and response spectra.

Finite Element Model is used for the RCS structure analysis.

Structural analysis for the normal operation conditions are performed in linear static analysis.

The dead weight, pressures and temperature is three additions for each operation condition are considered.

The seismic analysis performed using nuclear island component RCS model details the information described in Section 3.7.

The structural analysis for the postulated pipe breaks is also performed.

The effects on the postulated pipe break are the jet impingement and thrust subcompartment pressure, internal pressure into the blowdown and branch nozzle loads.

MEMBER SKILLMAN: Are those four bullets that you just presented breaks where LBB is not applied, jet impingement subcompartment, pressure blowdown and branch nozzle loads, the DFL loads that you show in your Table 3.9.2, the DFL loads are described as the dynamic fluid loads and occasional loads. Is that is what you are communicating here?

MR. P. HONGSUN: Yes.

MEMBER SKILLMAN: It appears that is what that is.

MR. P. HONGSUN: Yes.

MEMBER SKILLMAN: That's what that is?

MR. P. HONGSUN: Yes.

MEMBER SKILLMAN: Yes? Okay, thank you. It's in your table? The table is not specific. Here, you have been more specific and I presume there are still more that are not identified. But, that is a classification of your dynamic fluid loads.

Okay, thank you.

MR. P. HONGSUN: Next slide?

Flow-induced vibration assessment for the reactor internals is described in Subsection 3.9.2.3, 4 and 6 and technical records, especially vibration assessment programs are implemented for mechanical systems and components subjective to vibratory force.

The reactor internals important to safety are designed to accommodate steady state and transient vibratory loads throughout the service life.

The APR1400 reactor internals are classified as non-prototype Category 1 with Palo Verde Unit 1 of ES.

Because the APR1400 reactor internals are a substantially the same as Palo Verde Unit 1 with regard to arrangement, design, size and operating conditions.

Analysis program includes the calculation of hydraulic loads and dynamic response and a comparison between predicted and measures of responses.

Full inspections procedure will be conducting pre- and post- the hot functional tests.

For the dynamic system, unless it's for the reactor internals under faulted condition, the dynamic analysis for the reactor internals or core are performed to determine the maximum structural responses under pipe breaks and seismic loadings.

For excitation loads, pipe break blowdown loads and reactor vessel motion are applied to the structural analysis using Finite Element Method.

The analysis results for the core are provided to evaluate the structural integrity of the fuel assembly that is shown in DCD Section 4.2.

The analysis results for the faulted conditions show that the reactor internals meet the stress limits of the ASME Section 3 Subsection NG.

The dynamic analysis of the CEDM calculate the maximum structural responses in order to confirm the structural integrity of the pressure housings and scramability.

CHAIRMAN BALLINGER: Excuse me, I've been looking through the slides going forward and I'm trying to find a convenient place for a break and it doesn't seem like there actually is a convenient place for a break within the next 20 minutes or more.

So, what I would propose is that we do take a break now because this is at least a subdividing point and come back at 25 minutes until.

So, we'll adjourn until 25 minutes until.

MEMBER STETKAR: Recess.

CHAIRMAN BALLINGER: Recess, excuse me.

(Whereupon, the above-entitled matter went off the record at 10:20 a.m. and resumed at 10:35 a.m.)

CHAIRMAN BALLINGER: Okay, we're back in session.

MR. SISK: This is Rob Sisk, Westinghouse.

We wanted to make a clarification on the record to a question that was raised by Mr. Skillman relative to Table 3.9-1 and the text in 3.9-1.

MEMBER SKILLMAN: Yes?

MR. SISK: So, please, go ahead and

clarify.

MR. P. HONGSUN: DCD Section 3.9-1, Table 1, Service Level A condition is described that Service Level A condition consists of 14 event conditions. Okay?

And, Table 3.9-1, Level A Service Condition consists of 14 conditions. But, the number of events, nine, but we divided two or three events of Service Level 1, 2, 3. So, we have 14 events of Service Level A condition.

MEMBER SKILLMAN: Thank you for your explanation. I would like my question to remain on the record, because I have the table in front of me, and I see the nine events, and I see many more than 14 conditions.

MR. SISK: I think when you count the nine events, there are a couple of the events that had A, B, C, D.

MEMBER SKILLMAN: That's correct.

MR. SISK: And, when you count the events with the A, B, and C, D categorization, there's a total of 14.

MEMBER SKILLMAN: I'm pleased to stand corrected. Let me go back and take a look. I can see that what you have under the events are not only the subevents, but the total of the subevents. And, that would be a double count in each of the events and I had not appreciated that until now.

Thank you, that's good, thank you.

MR. P. HONGSUN: Thank you.

MEMBER SKILLMAN: Actually, I would like to comment that the reason I asked the question was in your behalf because it appears as though you might have been creating a great amount of analytical utilization for your cumulative utilization factor than you needed to. That's what I was thinking.

It seemed to me that you might have been double counting and I was curious why you might want to do that. That would certainly lead to conservatism, but that probably isn't something that you would want to do. That's why I asked the question.

Thank you.

MR. P. HONGSUN: Okay, thank you for your comments.

And, let's move -- let's start to read Section 3.9.3.

The Subsection describes the structural integrity of pressure retaining components, component supports and core support structures.

The loading conditions are categorized as design, Level A, B, C and D conditions.

Design pressure, temperature and other loading conditions are presented in this subsection.

By stress analysis and fatigue variations, it is confirmed that pressure components meet the stress level stress limits and the structure adequacy under the operation condition.

Pressure relieving devices are designed in accordance with the requirements of ASME Code Section III Appendix O.

The POSRVs are designed to provide overpressure protection for the RCS.

Pressure relieving devices for ASME Section III Class 2 systems are on the streamline and the containment isolation portion of the normal shutdown cooling system.

Pressure relieving devices for Class 3 systems are on heat exchanges, tanks and piping lines to prevent overpressurization of the components and systems.

The functional design and qualification of safety-related active components are performed in accordance with ASME QME-1.

The functional capability is confirmed by design analysis, inspection, testing and start up periodic in service testing.

Supports are designed and constructed in accordance with ASME Section III and ASME Code Cases.

Snubbers are minimized to the extent practical through the use of design optimization.

Reasonable assurance of snubber operability is provided incorporating analytical, design, installation, in service and verification criteria.

This subsection provides the information on design, functional requirements and operability assurance program for the control element drive mechanism.

The CDM for the APR1400 is basically the same as the system 80 CEDM. It has a lot of operation experiences in U.S. as well as Korea.

Since the CEDM pressure housing is a pressure boundary component, it is designed in accordance with ASME Section III Subsection NB.

Reactor pressure vessel internals refer to the core support structures and internal structures in accordance with ASME Section III Subsection NG.

The components of the reactor internals are divided into two major parts. These are core support barrel assembly on the left figure and the upper guide structure assembly on the right figure.

The core support barrel assembly consists of the core support barrel, the low support structure and ICI nozzle assembly and the core shrouds.

The upper guide structural assembly consists of the UGS barrel assembly and the inner barrel assembly.

The UGS barrel assembly consists of UGS support barrel, fuel alignment plate, UGS support plates and control element guide tube.

The reactor pressure vessel internals are classified in Safety Class 3 and Seismic Category 1.

This slide shows the hold down ring of reactor pressure vessel internals. As you can see, the detail drawing on the right figure, the hold down ring sits on the core support barrel of the flange.

The hold down ring provides extra force on the flange of the upper guide structure assembly and core support barrel assembly in order to minimize movement of the structures on the hydraulic forces.

And, that is designed to accommodate the differential thermal expansion between the reactor vessel and reactor internals in the vessel ledge region.

MEMBER SKILLMAN: Before you change this image, and Andy, thank you for including this in the presentation, KHNP, thank you for including this.

Please explain the physical status of the hold down ring before the reactor vessel head is placed on the reactor vessel lower flange.

Is the hold down ring in a configuration to be compressed or to be pre-loaded so that when the reactor vessel head is placed on the lower flange, the core support assembly is driven downward?

MR. P. HONGSUN: That's right.

MEMBER SKILLMAN: So, that's a spring that actually goes into bending and pushes it down?

MR. P. HONGSUN: That's right.

MR. SISK: I think it's above the washer.

MEMBER SKILLMAN: That's what I'm -- that's the question. That's the question.

MR. P. HONGSUN: That's right.

MEMBER SKILLMAN: Okay, now, is there a competing load path where the upper guide structure floats on the fuel springs and is, therefore, also driven down by the reactor vessel head to ensure that the fuel assemblies are clamped so they cannot react against reactor coolant pump flow?

MR. P. HONGSUN: Yes.

MEMBER SKILLMAN: They are? So, when you remove your reactor vessel head, does the core support assembly rise?

MR. P. HONGSUN: Yes.

MEMBER SKILLMAN: It does? By about how much?

MR. P. HONGSUN: I don't know.

MEMBER SKILLMAN: One centimeter? Half a centimeter?

MR. P. HONGSUN: I don't know exactly.

MEMBER SKILLMAN: But, you know that it rises?

MR. P. HONGSUN: Where it permits -- and Korean members --

MR. SUNG: This is Kikwang Sung from KEPCO E&C.

And, you asked -- do you ask the amount of the displacement of the core support barrel? Right?

MEMBER SKILLMAN: Not the core support barrel, it's of the upper guide assembly --

MR. SUNG: Upper guide --

MEMBER SKILLMAN: -- where it's up by the refuel springs?

MR. SUNG: And then, as far as I know, there is no relative displacement because the upper ledge and upper heads are clamped by using close head studs around the head. So, there is no vertical movement.

MEMBER SKILLMAN: Yes, I understand that.

Let's presume that you and I take a stud tensioner and we untension the head, take off the nuts --

MR. SUNG: Yes?

MEMBER SKILLMAN: -- reactor vessel head rises?

MR. SUNG: Rise.

MEMBER SKILLMAN: What pushes it up?

MR. SUNG: I am sorry that I don't think I can answer because --

MEMBER SKILLMAN: The answer is the springs on the fuel assemblies --

MR. SUNG: Yes.

MEMBER SKILLMAN: -- actually rise --

MR. SUNG: Yes, right.

MEMBER SKILLMAN: -- raise the upper structure.

MR. SUNG: Yes, right.

MEMBER SKILLMAN: Is that correct?

MR. SUNG: There are two kinds of hold down springs in the reactor internals. One is the hold down ring.

MEMBER SKILLMAN: The ring.

MR. SUNG: And, one spring is the --

MEMBER SKILLMAN: Fuel --

MR. SUNG: -- fuel assembly hold down ring. You are right.

MEMBER SKILLMAN: Okay. That answers my question. Where I was really going with this is to understand how the internals are, on the one hand, free to float until you place the reactor vessel head in place.

And, when you do place the reactor vessel head in place, how the internals are clamped and the fuel assemblies are set in their lower fittings.

And, you've explained it at least from a practical perspective. I do understand it.

Thank you, I'm done. Thanks.

MEMBER MARCH-LEUBA: I'm not -- can I ask a question?

This question he was drawing out for you, we've talked during LOCA analysis and somewhere it's up to 15 about this bypass flow from the upper plenum to the downcomer that relieves loop seals and other things that happen with LOCA.

Is that complicated path I see in this

figure where this bypass happen? Anybody know? You do?

MEMBER SKILLMAN: I'll leave it to KHNP first. Yes, if you look on the left image, if you look at the T hot outlet, if you look at that 41-inch diameter T hot exit, what happens is, as the reactor vessel internals come to temperature, the reactor vessel internals grow such that the outlet boss on the internals interface and clamp on to the machine outlet of the hotleg.

And, it's the closure of that 41-inch diameter circumference that seals against the leakage of which you --

MEMBER MARCH-LEUBA: So, it's not a complicated path on the right?

MEMBER SKILLMAN: Oh no, it's really the hotleg closure.

MEMBER MARCH-LEUBA: It's the hotleg closure, but it's being clamped very strongly by the upper head.

MEMBER SKILLMAN: Well, what happens is the upper head clamps the internals, but the internals grow and that fit up machining is a critical item for the design of the internals and the design of the reactor.

The final machining of the internals and the reactor vessel, yes, it's radial machining.

MEMBER MARCH-LEUBA: Okay, thank you.

CHAIRMAN BALLINGER: Before we keep -- are we through with that?

MEMBER SKILLMAN: Yes, I'm -- thank you. Thank you, I'm good.

CHAIRMAN BALLINGER: Pete and I have been going back and forth, if you look at your Table 3.9-1 which gives the transients, there's Event 2A and 2B which are daily load follow operation.

If you actually literally consider that to be daily and you look at 22,000 and 60 years, you don't come up with any kind of number that makes any sense.

MEMBER RICCARDELLA: No, 22,000 is like six years.

CHAIRMAN BALLINGER: Yes. But, if you look --

MEMBER SKILLMAN: Let me weigh in. That's why I asked the question earlier about normal Level 1A and 1B and Service Level A at 1.5 million.

CHAIRMAN BALLINGER: Yes.

MEMBER SKILLMAN: It appears as though the 1.5 million might include those first two Service

Level B events. That's why I asked that question.

CHAIRMAN BALLINGER: But, if you note in, fortunately or unfortunately, we've gotten both DCD Revision 0 and Revision 1.

If you look at Revision 1, there's a Note 2 which is at the end of that table which says, although the APR1400 will be operated as a base load plant, the effects of daily load follow operation are accounted for in the structural design and analysis of ASME Code Class 1 components, reactor internals, dah, dah, dah.

Somebody ought to fix the number because daily to me means daily. And, the numbers don't -- if you divide it up, it doesn't work.

MEMBER SKILLMAN: Well, I did that arithmetic and I got, for the 15 -- for the 1.5 million times during 60 years, I got 68 per day.

And so, that's getting pretty close to a couple times an hour. But they're small changes, yes. But that is very consistent with how a load follow plant would behave. You would hear the drives going click, click up, click, click down.

MEMBER RICCARDELLA: Yes, but wouldn't the load follow it one more than just like a 5 percent change? You know, you'd want to come down to maybe 50 percent at night, you know, like those other -- like those 22,000 cycles.

CHAIRMAN BALLINGER: But, I would note that in Rev 0 that note is not there.

MEMBER RICCARDELLA: What is it? Read the note again.

CHAIRMAN BALLINGER: The note says, although the APR1400 will be operated as a base load plant, which we always assumed, the effects of daily load follow operation are accounted for in the structural design and analysis of ASME Code Class 1 components, reactor internals and component supports.

But, that note is not in Rev 0.

MEMBER RICCARDELLA: Yes, right.

MEMBER SKILLMAN: Well, I can't fault KHNP for advertising a base load plant but also incorporating the cycles that could be required for a plant that might be sited in South Africa or in Japan or in Europe or Greenland or some place.

So, I certainly don't fault them for building that capability into the plant through their mechanical analysis.

MEMBER RICCARDELLA: Yes, but if it's only 22,000 cycles, it's not really daily for 60 years.

MEMBER STETKAR: It actually is if you

read in Rev 0 of the DCD how they define load cycle. It's one reduction per day which is 21,960 years. Sixty times 365 --

MEMBER SKILLMAN: Is that many cycles.

MEMBER STETKAR: -- is 21,900 -- wait a minute -- I can -- I have it here somewhere. Do you have a calculator? I did the math and they defined the load cycle as one cycle per day.

CHAIRMAN BALLINGER: This conversation is completely --

MEMBER STETKAR: This is what happens and this is on the public record, this is what happens when you get a bunch of doctorates together in a room. Eventually, you asymptotically approach something that might be similar to the correct answer.

CHAIRMAN BALLINGER: Or you can if you're stupid.

MEMBER RICCARDELLA: Understood, I apologize.

MEMBER SKILLMAN: But, thank you for your discussion --

MEMBER SKILLMAN: -- on the hold down ring. Thank you, I appreciate that. Okay, thank you.

MR. P. HONGSUN: Let's move on to Section 3.9.6.

This section is to discuss functional design qualification and in service testing program for pumps, valves and dynamic restraints.

IST program for ASME Code Section III, Class 1, 2 and 3 safety-related pumps, valves and dynamic restraint is developed.

The COL applicant will provide a full description of the IST program.

Functional qualifications are performed in accordance with ASME QME-1.

This is the end of Section 3.9. Mr. Choi will continue to present from Section 3.10.

MR CHOI: Hello, my name is Bosung Choi. I am an Equipment Qualification Engineer from KEPCO E&C to give presentation from Section 3.10 to Section 3.12.

Seismic qualification requirements for Seismic Category 1 instrumentation, electrical equipment and mechanical equipment are established to provide reasonable assurance of structure integrity and perform the designed safety-related functions or intended function under the postulated SSE in combination with other concurrent loading conditions identified in the equipment design specification.

Qualification standards used in the

seismic qualification include IEEE standards 344S modified by Reg Guide 1.100 for safety-related mechanical and electrical equipment and support.

And, IEEE Standard 323 is included for the equipment qualification in conjunction with the environmental qualification.

ASME Code Section III for structural integrity of safety related pressure boundary components are used for the seismic qualification.

And, ASME QME-1 2007 for qualification of active mechanical equipment is used for the criteria for the seismic qualification.

So the methods for qualifying mechanical and electrical equipment and instrumentation, testing is conducted for equipment that cannot be qualified with the analysis alone or equipment having components that potentially cause any malfunctions related to their intended functions.

Analysis without testing is acceptable only if structural integrity alone can assure the design intended function.

Combination of testing and analysis is utilized when the equipment cannot be practically qualified by analysis or testing alone.

Test or analysis is performed for the

supports of mechanical equipment, electrical equipment and instrumentation to assure structural integrity of the equipment being qualified.

When electrical equipment and instrumentation supports are qualified by testing, the components of the equipment are installed or dummies are installed.

The supports can be qualified by analysis when only the structural integrity of supports are interested.

For mechanical equipment support, qualification will be performed in accordance with ASME Code Section III.

When instrumentation line supports are qualified, the criteria from ASME Code Section III are applied. Subsection NF for equipment Class 1 and 2 are applicable.

And the seismic qualification file will include qualification methods used for equipment, test and analysis results, list of systems, equipment and equipment support structures, seismic qualifications, summary data sheets which summarizes the components qualification and seismic input requirements.

This will end the Section 3.10.

Section 3.11 is about the environmental

qualification of mechanical and electrical equipment.

Location of each equipment is specified as room numbers in DCD Table 3.11-2 and classified into mild or harsh condition based on the environmental conditions.

The environmental conditions to be considered in the equipment qualification are normal condition, AOOs, DBAs and post-DBA.

The conditions are as conditions as a result of analysis of certain events, environmental parameter values for each room are specified in Technical Report for equipment qualification program, as Table 3 in the EQP.

Environmental qualification of Class 1 equipment is consistent with the requirements of 10 CFR 50.49, 10 CFR 50 Appendix B, Reg Guide 1.89 and IEEE Standard 323.

For specific equipment, there are specific EQ Regulatory Guides and they endorse some IEEE standards. APR1400 uses these specific standards for certain equipment.

For example, IEEE 382 for valve actuator, IEEE 383 for cables and IEEE 649 for motor control center.

Test results of the qualification are

documented in auditable files in accordance with 10 CFR 50.49(j).

The vital instruments and equipment are served with 100 percent redundancy of HVAC unit.

Design basis that prevent the loss of safety-related ventilation are described in Section 6.4 and Section 9.4 of DCD.

The radiation qualification for individual safety-related components are developed based on the fact that the equipment will be qualified up to the time the equipment is required to remain functional following an accident and the limiting DBAs are LOCA in containment building, MSLB in main steam well house and FHA in fuel handling area.

The assumptions for determining normal and accident condition TIDs are as follows.

For estimation of the normal EQ TID values, the Microshield code was used. And, for the accident EQ TID values, RUNT-G code is used.

Assumptions and methodology for evaluating radiation doses for equipment qualification are based on Reg Guide 1.89 and 1.183 guidance.

According to the Reg Guide 1.89 and 1.183, the source term for the normal TIDs is from 1 percent failed fuel and the source term for the accident TIDs is from accident source term used for radiological consequence analysis in DCD Chapter 15.

Exposure time in radiation environment is 60 years of continuous operation with full power plus one year post-accident period.

Per Reg Guide 1.89 guidance, 10 percent margin is considered for uncertainty of tests.

Mechanical equipment can be divided into active and passive equipment. Active mechanical equipment has mechanical moving parts to achieve its safety function. For active mechanical equipment, environmental qualification focuses on non-metallic parts of the equipment in conformance with ASME QME-1 Appendix QR-B.

The main safety function of passive mechanical equipment is maintaining its structural integrity. As the structural integrity is achieved really by the analysis of the passive equipment.

Now, I'm going to move to Section 3.12, the piping design review.

This section provides the adequacy of the structural integrity as well as the functional capability of the safety-related piping system, piping components and their associated supports.

A graded approach is taken to the scope of piping systems and components design.

The scope of design for ASME Code Class 1 piping includes RCS main loops, pressurized surge line, DVI line and SC line.

And the scope of design for ASME Codes Class 2 and 3 piping includes main steam and main fuel piping.

Piping systems and supports are designed in accordance with 2007 Edition with 2008 Addenda of ASME Code Section III.

And, the piping analysis methods, the procedure used for analytical modeling selection of frequencies that being criteria combination of model responses is described.

For seismic analysis, response spectrum, time history and equivalent static load method is used.

The computer codes used for modeling is pipe stress ANSYS, RELAP5 and GTSTRUDL.

And the piping benchmark problems prescribed in NUREG/CR-1677 are used to validate the pipe stress computer program used in piping system analysis.

Based on methodology and the equations from the ASME Code, pipe stresses are calculated for various load combinations. The ASME Code includes design limits for design conditions such as Service Level A, B, C, D and testing.

The evaluation of the environmental fatigue for Class 1 piping is performed in accordance with Reg Guide 1.207.

Seismic Category 1 pipe supports are designed in accordance with ASME Section III Subsection NF for Service Levels A, B, C and D.

And, the acceptance limits of Appendix F of ASME Section III are used for Service Level D.

For non-seismic category pipe supports where supporting piping system is analyzed to ASME B31.1, the requirements of ASME B31.1 will be met where applicable.

Design of base plate and anchor bolt for piping support is in accordance with ACI 349-1 Appendix B and Reg Guide 1.199, Bulletin Letter 79-2.

This will end the presentation of Section 3.12.

Mr. Park will continue from Section 3.13.

MR. P. HONGSUN: Okay, let's move on to the Section 3.13.

This subsection provides the design features of the threaded fasteners.

In this slide, several design considerations on threaded fasteners are discussed.

At first, fastener materials are selected in accordance with ASME Section III NCA and NB/CD.

The fasteners are fabricated using the materials prescribed in ASME Section III or ASME Code Classes.

Boric acid corrosion and galvanic corrosion are also considered. Special processes and considers related --

MEMBER SKILLMAN: Excuse me. Please explain what you mean by that statement, prevention of boric acid corrosion, galvanic corrosion unless considered to be acceptable.

Are you suggesting that you are choosing materials that are immune to boric acid and to galvanic corrosion?

MR. P. HONGSUN: Yes, we considered prevention of boric acid, PAC and galvanic corrosion. So, we selected materials to consider this corrosion.

MEMBER SKILLMAN: What materials are immune to boric acid corrosion and to galvanic corrosion? I mean, some of these bolts, some of these studs are static at a very high stress intensity and small amounts of boric acid plus some amount of fluid that acts as an aqueous conduit has the capability to degrade the material.

So, I understand what I think you're trying to communicate. But, I'm curious from a practical perspective what it is you're going to do.

Any amount of electrolyte has the potential to shear off a bolt head, to shear off a nut. I've seen it happen.

CHAIRMAN BALLINGER: I think that there's a wording problem here. I think if you were -- the words unless considered to be acceptable, I'm not exactly sure what that means. I'm not sure if Dick knows what it means either.

MEMBER SKILLMAN: No, that's why I'm asking. What I do know is that, for the pressurized water reactors, at least that I'm aware of, following the Davis-Besse incident, there was a massive attack on boric acid leakage.

And, every licensee had to prove that the licensee was ensuring that boric acid leakage was prevented or, if it was discovered, that whatever equipment the boric acid attached itself to was not degraded.

But, the thrust of this is under material selection, so I'm wondering if you have some very superior material that is immune to galvanic corrosion or to boric acid corrosion?

CHAIRMAN BALLINGER: The wording is just very murky.

MR. SISK: Yes, this is Rob Sisk, Westinghouse.

I certainly agree with you. And, I believe what you're seeing there is a wording issue. But, I think the message here is that the material selection takes into consideration the boric acid, galvanic corrosions.

And, I think, Dick, as you pointed out, there's nothing that's totally immune, but you have to be aware of it and factor that into an incremented maintenance program, any sort of program for that equipment as you go forward.

So, I think the issue here is to make sure that we fully account for these factors in the selection of the material and their application.

MEMBER SKILLMAN: That's fine. I understand. Thank you.

MR. P. HONGSUN: Let's us go -- let me go.

And, the special processes and controls related to material properties are in accordance with Section II and III and threaded fasteners are cleaned in accordance with Regulatory Guide 1.28.

During fabrication, fasteners are inspected in accordance with ASME Section III and B, C or D.

Lubricants are selected in accordance with the guidance provided in NUREG report. Acceptable lubricants are Loctite N-5000, Neolube and Never Seez Pure Nickel Special Nuclear Grade.

However, molybdenum disulfide is not allowed for use in any circumstances.

Ferritic fasteners are tested in accordance with Section III and B, C or D and 10 CFR 50 Appendix G.

In addition, all CMTRs are controlled, maintained and stored in accordance with the quality assurance program.

Pre-service and in-service inspection follows the relevant requirements of ASME Section XI. COL applicant is to submit the inspection programs.

This is the end of Section 3.13 and Chapter 3 and the last of presentation materials, RAI Summaries and Status, Open Items and COL Items are summarized for your information.

Thank you for your listening.

CHAIRMAN BALLINGER: Is it your intention to stop here?

MR. SISK: This is Rob Sisk, Westinghouse.

Yes, as mentioned, the open items and the COL items in the back are provided for your information. Given the brevity of the time, we had to cover the chapter. We weren't going to try to cover all the open items. So, we're finished with Chapter 3 at this time unless there are additional questions from the Committee.

CHAIRMAN BALLINGER: Now, I guess I'm in a quandary. I sort of thought we would go longer, so I didn't ask the staff if they were ready to start. So, I'm assuming they're not ready to start. Okay?

So, that solves that problem.

Which means that we don't have much of a choice except to --

MEMBER REMPE: Before we bang the gavel, could we start earlier than the plan?

CHAIRMAN BALLINGER: They just nodded no.

MR. WARD: We could start earlier, we can't start now, but we could probably --

CHAIRMAN BALLINGER: Start at 12:30?

MR. WARD: Yes, we'll round up the staff and we can start earlier.

MEMBER STETKAR: Just as long as -- just

announce it.

CHAIRMAN BALLINGER: Okay, all right, option A.

MEMBER STETKAR: Because we're on the public record, so if anyone is listening in they know when to tune in again.

CHAIRMAN BALLINGER: Okay. After some discussion, we will recess until 12:30 at which time we'll pick up the staff's presentation.

(Whereupon, the above-entitled matter went off the record at 11:16 a.m. and resumed at 12:30 p.m.)

CHAIRMAN BALLINGER: Okay, we're back in session with the staff's presentation, and I'm not sure who's going to go first, but whoever it is, go first.

MS. TERRY: Good. Hi, good afternoon. My name is Tomeka Terry, and I am the Chapter Project Manager for APR1400 Design Certification Application Review for Chapter 3.

This slide provides a list of the technical staff who will be presenting today -- this afternoon, ACRS.

In Section 3.1 through Section 3.6.1, 3.6.4 and 3.13 were no specific issues in these sections. However, if ACRS Members have any questions, the staff will be happy to answer your questions.

Next slide?

Okay, during the design certification review, the staff conducted pre-application QA audits and performed audits at the applicant facility.

The staff also held biweekly public meetings with the applicant to resolve technical issues and to improve RAIs which the staff issued several RAIs to the applicant.

Also, the staff incorporate lessons learned from previous design certification applications to ensure consistency with other designs certifications, too.

Now, I will turn it over to Robert Rivera who will be presenting seismic parameters and analysis SER Section 3.7.

MR. ROCHE RIVERA: Thank you, Tomeka.

Good afternoon, my name is Robert Roche-Rivera. I'm a Structural Engineer in the Structural Engineering Branch in the Office of New Reactors.

My colleague, Jinsuo Nie and I were the reviewers of DCD Section 3.7 related to the applicant's seismic parameters and analysis.

Specifically, DCD Section 3.7.1 over seismic parameters, Section 3.7.2 covers seismic system analysis. Section 3.7.3 covers seismic subsystem analysis.

In a moment, Jinsuo will present review topics related to 3.7.1 and 3.7.3 and I'll present review topics related to Section 3.7.2.

I would also like to recognize the support from BNL, Brookhaven National Laboratory who supported our review of the APR1400 application.

On the phone, we have -- we should have Joe Bragerman. Joe, are you there?

(No response.)

MR. ROCHE RIVERA: Okay, he should join momentarily and then here in the audience, we have Dr. Carl Constantino and Dr. Thomas Houston.

So, next slide, please?

So, this slide provides an overview of our review of the applicant's seismic parameters and analysis.

As part of our review process, we reviewed DCD Section 3.7, and this is 3.7(a) which is related to the applicant's full structural interaction analysis.

Appendix 3.7(b) which describes the

applicant's evaluation for high frequency seismic input as well as six referenced technical reports supporting all these evaluations.

We held biweekly public meetings with the applicant to discuss technical issues. We preformed independent confirmatory analysis of the acceleration tank histories used for input to the seismic analysis.

We conducted a seismic audit, seismic design audit. We confirmed consistency with the staff guidance and, with the exception of one open item which is related to the applicant's evaluation for high frequency seismic input, we concluded that the seismic design parameters and seismic analysis procedures and criteria delineated by the applicant provides an acceptable basis for the seismic design.

I will now turn over the presentation to my college Jinsuo Nie.

MR. NIE: Thanks, Robert.

Good afternoon. My name is Jinsuo Nie. I am a Structural Engineer.

I was responsible for reviewing two areas, first the seismic design parameters and the other one is seismic subsystem analysis.

All identified technical issues in these two areas have been resolved and all RAIs are confirmatory.

For my presentation, I selected three examples that I hope are of interest to the Committee.

These examples are in the area of seismic design parameters.

For seismic subsystem analysis, the DCD covers generic methodologies and the only physical component in this area, part of the piping, conduit and tunnels, concrete damps and above ground tanks are treated as COL information items.

I'll now present the information there. If there are any questions, I can answer.

Next, please?

My first example is about the evaluation of CSDRS consistent foundation level response spectra in the free field or $CSDRS_{ff}$.

Appendix S to 10 CFR Part 50 requires $CSDRS_{ff}$ to bound the minimum required response spectra for MRRS. MRRS for APR1400 is defined as the CSDRS anchored to 0.1g in the horizontal directions.

The applicant initially used the envelope of the $CSDRS_{ff}$ for all soil profiles to compare with MRRS.

Since each soil profile can be evaluated at the plant site, and the staff's request, the applicant compared $CSDRS_{ff}$ for each soil profile and determined that they bound MRRS.

$CSDRS_{ff}$ was found to have large dips for two soil profiles, S6 and S7. The applicant explained that these dips were caused by a soil layer interface right about the foundation level for these two soil profiles.

However, the transfer functions providing RAI response did not appear to explain these dips. During the audit Robert just mentioned, the staff reviewed the applicant input file for said response analysis and found the transfer functions were not from the ground surface to the foundation level.

But, instead, they found the bedrock to the foundation level.

And, the applicant later provided the transfer functions from the ground surface to the foundation level which showed dips consistent with the dips in $CSDRS_{ff}$. So this can resolve this particular technical issue.

Next, please?

My second example is about the development of targeted power spectra density function or PSD.

SRP 3.7.1 Option 1, Approach 1 makes provisions to check the PSDs of the design time histories against targeted PSDs as a secondary check.

The applicant has specified the targeted PSD for the horizontal CSDRS below 9 hertz as the one for the Reg Guide 1.60 horizontal spectra.

The vertical target PSD was based on one time scaling of the horizontal targeted PSD. This measure may not produce a targets PSD compatible with the CSDRS.

In the RAI response, the applicant showed that these targeted PSDs were in general agreement with those developed based on 30 simulated time histories.

The staff's confirmatory analysis showed that its targeted PSDs were higher in some frequency ranges than the applicant's targeted PSD.

However, the applicant's minimum targeted PSDs which were actually used in assessing power sufficiency of the design time histories were very close to or higher than the staff's minimum targeted PSDs.

The reason was that the applicant used a factor of 0.8 which is slightly higher than the 0.7 specified in SRP 3.7.1 Appendix B.

Just for background, a higher minimum targeted PSD means a more conservative PSD check.

Next slide, please?

My last example is about the seed records and design time histories.

The staff conducted a confirmatory analysis of the seed records because the staff -- because those records do not show high frequency contents and the staff had a concern how they could affect the design time histories that have an accuracy frequency of 100 hertz.

It was found out that Fourier phase spectra of the seed records were cyclic constant or with a gap in some frequency ranges.

However, the design time histories were not found to have this unrealistic features and they have high frequency contents.

The staff confirmatory analysis also identified the low pass filters were applied to the design time histories with corner frequencies lower than 50 hertz.

The applicant explained that the low pass filters were used to remove artificial high frequency contents due to baseline correction and clipping in the spectra matching process.

The effect of the low pass filters were determined to be insignificant because the low corner frequencies of only 48 hertz well within the 10 hertz frequency window for PSD modeling and affected hertz.

The last point on this slide is that to estimate the PSD of the design time histories, the applicant initially used a method that was not consistent with the SRP guidance.

In some cases, that method may overestimate the PSD for frequencies with small amplitudes.

The applicant's comparison of the two methods show that the generally produced comparable PSD estimates, in some cases, they differ by a factor of about three at some frequencies.

Both the applicant's revised calculation and the staff confirmatory analysis confirmed that the PSD of the design time histories are higher than the minimum targeted PSD which we just talked about in the previous slide, shown in other design time history has sufficient power.

This completes my presentation and I will be glad to answer any questions you may have.

So, we're good? Okay, I now will turn over my seat to David Heeszel and the turn to my colleague, Robert who will talk about the seismic system analysis.

Thank you.

MR. ROCHE-RIVERA: Okay, thank you Jinsuo.

Good afternoon, again.

I was the reviewer for DCD Section 3.7.2 which documented the applicant's seismic system analysis.

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

In the next few slides, I'll present two topics that are related to the review areas within DCD Section 3.7.2.

Next slide, please?

The first topic is on soil structure interaction sensitivity studies.

I think maybe the previous slide, there we go, yes, there we go. Thank you.

So, in accordance with the acceptance criteria in SRP 3.7.2, Roman Numeral II, IV, sensitivity studies are performed to assist in evaluating the adequacy of the seismic response obtained from linear SSI analysis methods.

For example, studies are performed to

verify whether nonlinear effects associated with the separation between soil and sidewalls that imbedded or foundation uplift would adversely affect the results that are obtained from SSI linear analysis.

So, the staff review found that the original submittal lacked the performance of sensitivity studies.

Also, while the original submittal included a study of the basement uplift, such study did not appear to be consistent with the guidance in the SRP.

Now, to address these issues, the applicant revised its basement uplift study and performed additional studies as follows and are delineated in the slide.

So, the applicant performed a study of the separation of soil from the sidewalls that are embedded.

As part of its study, the applicant performed SSI analysis of the nuclear island with soft soil and hard rock soil profiles with modifications to allow or permit separation between the soil and the walls.

The applicant compared the in structure response spectra obtained from this study with the corresponding in structure response spectra obtained from the fixed space analysis case for which the separation of the soil and the wall is not a concern. In other words, it's not applicable.

Also, this fixed-base case is part of the design envelope that forms the envelope in structure response spectra for the APR1400 structures.

So, what this comparison showed was that the in structure response spectra from the fixed-base analysis bound the in structure response spectra from the analysis that allowed separation between the soil and the sidewalls that are embedded.

And, on this basis, the staff found that the assumption of no soil separation in the design basis linear analysis to be acceptable.

The applicant also performed analysis cases for several values of concentration and demonstrated that none of the variables exist in the design basis SSI analysis to the Poisson's Ratio values that were used in the design basis analysis.

And, lastly, the applicant revised its study of basement uplift for consistency with the SRP guidance and demonstrated that the contact ratios between the basement and the subgrade media below it are greater than 80 percent which meets the SRP criteria in Section 3.7.2 Roman Numeral II, IV for acceptability of the linear analysis response.

Next slide, please?

The second topic that I would like to present, it's on structural soil structural interaction analysis.

So, in accordance with the acceptance criteria in the SRP, again, Section 3.7.2, the significance of SSSI analysis has to be evaluated and addressed if significant in the design -- in the seismic design.

So, while the original submittal included a SSSI evaluation, such evaluation assumed the surface founded the structure and did not consider the pressures on the exterior below grade walls due to the SSSI effects.

To address this issue, the applicant performed a SSSI analysis based on an embedded foundation configuration. Or, in other words, considering embedment.

So, the results of this evaluation showed that the lateral soil pressures computed from the SSSI as well as the SSI analysis were higher than the dynamic soil pressure that was originally used in the design of the exterior below grade walls in the auxiliary building and the diesel fuel oil tank room.

To address the exceedance of the original dynamic soil pressure, the applicant reevaluated the structural design of exterior below grade walls to consider the calculated maximum lateral soil pressures from the SSSI and SSI analysis.

The structural and soil aspect of the structural design of the below grade walls will be discussed in more details in a few moment by my colleague Vaughn Thomas.

So, this right now, completes my portion of the presentation. If there's no questions, then I will turn over the presentation to Vaughn Thomas.

MR. THOMAS: Okay, good afternoon and thanks, Robert.

Good afternoon, my name is Vaughn Thomas. I'm Structural Engineer in the Office of New Reactors.

In the next few slides, I'm going to present some of the key technical issues identified during the review of Section 8.1 through 3.8.4. 3.8.1 has to do with containment, the concrete containment at 3.8.2.

They do have a steel containment, but they are components that are -- that are not backed by concrete, for example.

They have a fuel line, but I'll talk about that in my next few slides.

Some of the issues that are associated with the view that were identified were, we talked about the hydrogen duration pressure load in the design of the concrete containment, the ultimate pressure capacity of the concrete containment, the leak chase channels in the containment internal structures and the dynamic lateral of pressure of the below grade walls as Robert alluded to.

Next slide?

The staff focused its structural review on the design approach and methodology to ensure that the applicant's design of the Cat 1 structures are reasonable and acceptable in terms of the scope level of details and technical adequacy of the information provided.

The staff performed the review in accordance with the Agency regulatory requirements and staff guidance.

The staff held numerous biweekly meetings with the applicant to discuss technical issues and resolution of RAIs.

The staff also examined and compared

applicant's results to the applicable code allowable.

The staff confirmed the applicant's method for demonstrating the design adequacy of the structures are consistent with Agency regulatory requirements.

Next slide, please?

The first topic of interest is the hydrogen duration pressure load using the design of the concrete containment.

In reviewing the load and load combinations associated with the design of the concrete containment, the staff noticed that the DCD did not describe the design and analysis procedures and the acceptable acceptance criteria for the hydrogen generation pressure load due to fuel clad and water reaction.

As a result, the staff requested additional information to ensure that the containment structural integrity is maintained when subjected to hydrogen generation pressure loads resulting from fuel cladding and water interaction.

To resolve this issue, the applicant performed additional calculation and sensitivity analysis of the 3D Finite Element Model of the containment structure.

The applicant's result demonstrated that the liner strains are below the ASME Code allowable limits and that the rebar and tendons remain in elastic range.

The applicant also demonstrated that the structural integrity of the containment subject or hydrogen pressure load meets the NRC regulatory requirements.

The applicant provided model copies that describe the design analysis approach and the acceptance criteria for the hydrogen pressure load.

The staff concludes that the information provided by the applicant demonstrated the containment structural integrity is adequately maintained when subject to hydrogen generated pressure loads resulting from fuel cladding and water interaction. Therefore, this issue is resolved and is being tracked as a confirmatory item.

Next slide, please?

MEMBER SKILLMAN: Vaughn, let me ask this question, please?

MR. THOMAS: Yes?

MEMBER SKILLMAN: As I read the safety evaluation, it's page 3-205, the staff communicates with specific regard to the containment pressure.

Ultimate pressure level -- at the ultimate pressure level the maximum strains of the rebar and tendons do not reach the allowable limit strain values. In addition, with regard to the punching shear, local failure of concrete near discontinuities such as the equipment hatch, the shear capacity of shear rebar exceeds the shear force corresponding to the ultimate pressure level.

However, the concrete shear strength is conservatively neglected.

And, as I interpret that, that tells me that you've placed all of the load on a combination of the rebar and the tendons. And, although the concrete may be able to take a huge amount of the load, it is conservatively ignored.

Hence, there is conservatism in this design that is really not being taken credit for. Is that accurate?

MR. THOMAS: Yes. Yes, sir.

MEMBER SKILLMAN: Thank you.

MR. THOMAS: The second topic of interest is the ultimate capacity of the concrete containment.

In reviewing the design and analysis procedures for the concrete containment, the staff noticed that the applicant's approach and acceptance criteria used to calculate the UPC of the containment does not accurately to be in accordance with the approach described in our regulatory position of Reg Guide 1.216 which is the containment structural integrity evaluation for internal pressure loadings above design basis pressure.

As a result, the staff requested additional information in order to better understand the applicant's design and analysis approach for determining the ultimate capacity of the concrete containment.

To resolve this issue, the applicant is committed to using Reg Guide 1.216 design and acceptance criteria for determining the UPC of the concrete containment.

The applicant also performed additional calculations and sensitivity analysis of the 3D Finite Element Model of the containment structure.

The applicant demonstrated through its results that the rebar, tendon and liner streams at the ultimate pressure level are below the stream limits.

The applicant also included a COL item which requires the COL applicant to provide the detail design results and evaluation of the penetrations and it shows equipment like equipment hatch, personnel airlocks and so on.

The staff concludes that the information provided by the applicant demonstrates the design and analysis procedures for determining the ultimate capacity of the concrete containment is acceptable. Therefore, this issue is resolved and is being tracked as a confirmatory item.

Next slide, please?

For the third topic of interest which is the leak chase tunnels as part of the design of the IRWST, the staff reviewed the components associated with the APR1400 containment internal structures, particularly the IRWST and noticed that the applicant did not provide a description or associated design details of the leak chase channels in the IRWST.

The leak chase channel collection system intends to prevent leakage of the braided water from the IRWST to its running concrete and containment liner below.

As a result, the staff requested additional information in order to adequately review the design and analysis procedures of the IRWST, including the leak chase channels and the stainless steel liner plate.

To resolve this issue, the applicant is committed to using a leak chase channel to the extent to monitor potential leakage of water from the IRWST.

The applicant provided backup copies of applicable sections of the DCD that includes the description of the leak chase channels collection system that will be used in the design of the IRWST.

The applicant included a COL item for monitoring and inspecting of the leak chase channels collection system.

The staff concludes that the information provided by the applicant demonstrates that the IRWST design and analysis procedure is adequate. Therefore, this issue is resolved and is being tracked as a confirmatory item.

Next slide, please?

For the final topic of interest, the evaluation of dynamic lateral earth pressures acting on the embedded walls like Robert talked about a little bit earlier.

The staff reviewed the design and analysis procedures in Section 3.4 and noticed that the applicant did not adequately describe the approach for developing the dynamic earth pressure loads that are considered in the analysis and design of the embedded walls.

As a result, the staff requested additional information related to the design and analysis procedures for the below grade walls for the Cat 1 structures considering the governing dynamic lateral earth pressures.

To resolve this issue, the applicant performed a sensitivity analysis which showed that, for the aux building and the diesel fuel oil tank structures, dynamic earth pressure determined from the SSI and the SSSI and analysis results were higher than those calculated in accordance with ASCE-498 method which was previously used for the design of the imbedded walls.

As a result, the applicant revised its structural analysis of the aux building and the DFOT and used the dynamic earth pressure obtained from the SSI and SSSI analysis as the governing dynamic earth pressure.

The increase in the dynamic earth pressure led to revising the member forces of the exterior walls.

The applicant provided markup copies of applicable sections of the DCD that reflect those changes.

The staff concludes that the information provided by the applicant adequately addressed the design and analysis of the below grade walls for the Cat 1 structures. Therefore, this issue is resolved and is being tracked as a confirmatory item.

And, this concludes my presentation. And, if you have any questions, I'll be happy to address them.

And, if not, I'll just turn to my colleague, Ata Istar to present the reapplied foundation.

MR. ISTAR: Good afternoon. My name is Ata Istar, Structural Engineer at Office of New Reactors.

And, I reviewed Section 3.8.5 foundations and I'll present the following selected examples during the review.

Could you turn to --

First, tendon gallery. The tendon gallery is the integral part of the nuclear island foundation of the APR1400 design.

However, the applicant did not describe the tendon gallery and the analysis and the design approaches for the tendon gallery in the DCD.

According to the SRP and areas of review, any unique design features that appear in the load pad needs to be describe which includes any safety-related function that the tendon gallery may have as part of the foundation and the pre-stress containment.

The applicant responded and provided a description of the tendon gallery and the tendon gallery was included in the analysis and design as part of the nuclear island common basement.

Therefore, this item is resolved and is considered as a confirmatory item.

Next slide, please?

A waterproofing membrane, the applicant did not provide any description whether waterproofing membrane used in the DCD.

Accordingly, the SRP in the areas overview, if waterproofing membranes are used, their effects on the shear resistance on the foundations needs to be addressed and the appropriate coefficient of friction for the waterproofing membrane needs to be considered when performing the stability evaluation.

The applicant responded that the waterproofing membrane will be used for the exterior below grade horizontal and vertical surfaces of the structure and nuclear island pad's basement in the APR1400 design.

The applicant also included the COL item 3.8.13 to verify the coefficient of friction between the lean concrete and the waterproofing membrane is greater than 0.55 used in the design.

Therefore, this item is resolved and considered as a confirmatory item as well.

Next slide, please?

Construction sequence and differential settlements. The applicant did not include this super structure or the reactor containment building and auxiliary building into the construction sequence evaluation and did not clearly determine the settlement types of maximum vertical settlement, tilt settlement, differential settlement between the buildings and angle of distortion throughout the nuclear island foundation.

According to the SRP in areas of review, loads that are induced by the construction sequence and differential settlements needs to be addressed.

However, currently, this issue is being addressed by the applicant and it's considered to be an open item.

This concludes my presentation. If you have any questions, I will be glad to answer.

MR. HEESZEL: Good afternoon, I'm David

Heeszel, a geophysicist in NRR. I'll be presenting the seismic instrumentation portion of the presentation.

The staff found that the applicant has adequately described the location of seismic instrumentation and the plans the COL applicants will have to develop for locating the instruments.

In addition, the applicant has cited appropriate regulatory guides that the COL applicants will have to follow for determining the exceedance of an OBE and plant inspection or restart following a seismic event.

And that concludes my presentation.

MS. TERRY: So, should we take a break or --

CHAIRMAN BALLINGER: No, we can keep going.

MS. TERRY: We should go? Okay.

So, let me change panels.

CHAIRMAN BALLINGER: We're off the record for a minute.

(Whereupon, the above-entitled matter went off the record at 1:04 p.m. and resumed at 1:11 p.m.)

CHAIRMAN BALLINGER: We're back in session.

MR. WONG: My name is Yuken Wong. I'm the Mechanical Engineer Branch. I'm the reviewer for Section 3.10, seismic and dynamic testing of qualification of equipment. I will also present Sections 3.9.2, 3.9.5 and 3.12.

The reviewer for these sections can answer any questions if needed.

Section 3.9.2, dynamic testing and analysis of systems structures and components.

Next slide, please?

The staff reviewed the methodology testing procedures, inspection program and dynamic analysis to ensure the structural integrity and functionality of piping systems, mechanical equipment and their supports under regulatory ruling.

Especially the staff reviewed the following six areas, piping vibration, thermal expansion, dynamic effects testing during initial start testing for ASME order and pressure vessel code Section III 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-operation of flow induced vibration testing of reactor internals, dynamic analysis of the reactor internals under faulted conditions.

Lastly, correlation of reactor internals and vibration test results with analytical results.

Next slide, please?

The staff also reviewed the comprehensive vibration assessment program, or CVAP report, for APR1400 steam generator and reactor internal design in comparison with System 80 design.

The Palo Verde Unit 1 reactor internal design is designated as the prototype and APR1400 as the non-prototype Category 1.

The staff issued an RAI to request the basis for using 33 hertz instead of the 0 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 applicant revised the DCD to use 50 hertz. The staff finds the response acceptable.

The staff issued an RAI to determine whether the fluid structure interaction and sloshing effects are included in the tank analysis.

The applicant responded that hydrodynamic forces exerted by the fluid on the tank walls are included in the analysis.

The staff finds the response acceptable.

The staff requested the benchmarking information for the DPVIB computer code used for the pump pulsation pressure analysis.

The applicant provided the verification and validation information in Revision 1 of the code.

The staff finds the response acceptable.

Next slide, please?

Section 3.5, reactor pressure vessel internals. The staff reviewed the arrangement of reactor internals, their functions, flow paths through the reactor vessel and design criteria.

Compared the APR1400 reactor internal design and CE System 80 reactor internal design, i.e., Palo Verde Units 1, 2 and 3 as their designs are similar.

The staff held public meetings with the applicant early in the design review process. Many of these issues were addressed by the applicant early. The remaining and additional issues were raised through RAIs.

The staff verified the core support structures and internal structures are designed and constructed in accordance with ASME order and pressure vessel code Subsection NG.

Next slide, please?

One open item is about the control element guide tube structure integrity. KHNP has since provided calculations to show the guide tube structure integrity can be maintained during a safe shutdown earthquake.

The staff finds the analysis acceptable.

The staff also request that KHNP to look into the operational history from operating plants with the System 80+ design.

KHNP investigated, Young Gwang Unit 2 since July 1995 and found no failure data on any reactor internals or guide tubes.

The staff asked KHNP to check Kori and Palo Verde as well. The status is ongoing.

The other open item is regarding the static O-ring at the seal table for the in-core instrumentation support system.

KHNP provided test data to validate the O-ring design.

The staff finds the response acceptable.

The last open item is about the seismic category for reactor internals.

KHNP provided clarification that all reactor internals including internal structures are classified as Seismic Category 1.

The staff finds the response acceptable.

Next slide, please?

Section 3.10, seismic and dynamic qualification of equipment.

Next slide?

The staff reviewed DCD Section 3.10, 3.7, B.7.4 and technical reports related to the equipment qualification, verified that equipment seismic qualifications standards and methods are in accordance with Reg Guide 1.100, IEEE 344 and ASME QME-1.

Verified procedures to evaluate the effects of hot rock high frequency, or HRHF, response spectra on high frequency sensitive equipment.

For new equipment, qualification will envelop certified as seismic design response spectra, or CSDRS and HRHF response spectra.

For equipment that has undergone prior qualification, the applicant will verify that test results envelop CSDRS and HRHF response spectra.

The prior test results do not envelop the HRHF response spectra. The applicant will perform screening tests using required response spectra enveloping HRHF response spectra.

The staff finds the response acceptable.

The staff conducted an audit of

procurement and design specifications to verify seismic qualification methodologies are in accordance with the DCD.

Next slide, please?

There is an open item related to the audit findings. The staff conducted a follow up audit and verified that the applicant's resolutions of the staff audit findings are acceptable.

The applicant will update a procurement specifications to address the staff audit findings.

Next slide, please?

MEMBER SKILLMAN: May I ask a question, please? And, it's on Section 3.10, if you go back a slide.

You have a COL item, it's COL 3.10(1), it's on your safety evaluation page 3-411.

And, the way this COL item is worded is as follows, the COL applicant is to provide documentation that the designs of Seismic Category 1 SSCs are analyzed for OBE if OBE is higher than one-third SSE.

I understand the concept that if the OBE is higher than one-third of the SSE that you need an analysis. But, I thought that this application established that, by definition, the OBE is one-third the SSE.

So, I'm curious where there would be an instance where a potential COL applicant would have a higher OBE than one-third of the SSE?

MR. WONG: You're correct. The DCD does establish the OBE equals one-third the SSE.

This COL item, it gives the COL applicant the option to set their OBE level. If they do not taking departure from the DCD, then there's no need to take action on this COL item.

MEMBER SKILLMAN: Does the staff anticipate that a COL applicant would take a departure?

MR. WONG: I do not expect the COL applicant would take a departure.

MEMBER SKILLMAN: Yes, I would not think so. But, that's why I'm asking the question. I can understand the notion that if somebody wanted to have a higher OBE then they would have to justify it which is what this question is pointing to.

But, I'm -- I guess I'm asking why the staff would anticipate that if the application for the design certification establishes that the OBE is one-third the SSE?

MR. WONG: Yes, I do not expect the applicant, COL applicant, would set a different OBE level.

MEMBER SKILLMAN: Thank you.

MR. WONG: Section 3.12 ASME Code Class 1, 2 and 3 piping systems and associated supports.

The staff confirmed that APR1400 piping and support analysis is in accordance with NRC guidance, conducted an audit of piping stress analysis and support designs to confirm consistency with the DCD, held public meetings with the applicant to discuss technical issues, issue RAIs that led to proposed DCD markup changes, revisions to the technical supports and added piping analysis.

Identified the environmental to assess the fatigue for the reactor coolant loop piping had not been performed.

The applicant has since completed the analysis and is currently under staff review.

The staff questioned the seismic analysis approach used. The applicant revised the calculation and the staff finds the revised calculation acceptable.

Next slide, please?

The applicant revised the nonlinear analysis using the time history methods and the staff finds it acceptable.

The HRHF response spectrum analysis, seismic analysis of piping within the scope of greater approach has been omitted.

The applicant has since completed the analysis and it will be included in a revision to technical report.

The evaluation of effects of HRHF response spectra on SSCs. This report is currently under review by the staff.

The staff requested the information regarding the structural integrity or piping and pipe supports that could be impacted by vibration and water hammer which could potentially originate from the operation and safety injection tank, operation of the safety injection tank and its fluidic device.

The applicant's response is pending.

This concludes --

MEMBER SKILLMAN: Is there a date for that product from the applicant? Because, I would like to see it and perhaps some of my colleagues would like to see that.

MR. WONG: Okay. It's the review of Section 3.10.

MR. TSIRIGOTIS: My name is Alexander Tsirigotis, I'm a Mechanical Engineer in the

Mechanical Engineering Branch.

We engaged the applicant with questions about the operation of the safety injection tank and its related device.

This was like two or three months ago. Since then, the applicant provided the write-up which I received today and I haven't had enough time to complete my review.

In addition, two weeks ago, one or two weeks ago, we prepared the official Request for Additional Information on this issue and ended up sending it out to them.

MEMBER SKILLMAN: Thank you, thank you.

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

The topics I'm going to talk about is determination of rupture location and dynamic effects associated with postulate rupture piping.

The review goal is to ensure that SSCs important to safety will accommodate or protect against the effects of postulate pipe failure.

The review approach -- next slide.

The staff reviewed the applicant's criteria used to define the pipe break and leakage crack location and consideration which includes circumferential break, longitudinal break and through leakage crack for both high energy line and also considered the model image piping.

The staff also reviewed the outline of the information which will be included in the pipe break hazard analysis summary report. From now on, I'll refer as PRHA report.

And, the staff has requested the applicant to submit a PRHA summary report upon their completion.

The staff also reviewed the applicant's methodology for addressing the non-potential non-conservatism of ANS 58.2 standard related to the dynamic jet modeling.

As you may know, the ANS 58.2 standard jet modeling has been commonly used by nuclear industry. However, following the interaction with ACRS Members during the general issue 191 review, the staff determined that there are four potential non-conservatism in the standard jet modeling.

First, the standard does not consider the brass whip effect.

Second, the standard assumes at universal jet expansion configuration. However, the characteristic of the jet expansion are not universal and are highly depends on the ratio of the fluid condition in the pipe that postulate failure to the ambient condition.

And, also, some testing results has shown that for the steam jet, the jet can travel much longer than the distance that would be determined by the standard model.

And, the third issue is that related to the jet -- the pressure distribution within the jet plume. For certain cases, the standard assumes that the peak pressure that can occur along the jet center line, that's valid for the area that close to the break exit.

However, four feet from the jet often the jet pressure will peak along the outer edge of the jet.

And, lastly, the standard does not consider the potential feedback amplification and the resonance due to the line break structure in case that synchronize the jet time scale.

So, in light of those potential non-conservatisms, the staff request applicant, because, in the DCD, it refers to the ANS 58.2 standard, so request applicant address those potential non-conservatisms.

And, in response, the applicant submit a design specific technical report that address those four issues.

And, based on the staff review of both the DCD and the technical report, first, the staff found out the applicant's criteria for defining the break and the leakage crack location and the configuration are consistent with the staff guideline.

However, the application of the break exclusion area, the applicant's area of break exclusion is beyond what the break potential position described for the containment penetration.

Assuming now, the staff guideline for containment penetration as the containment wall and that include the both input and output containment isolation valve.

For this region, the staff guidance describes certain design provisions, also the inspect requirement.

The provision of those additional design requirements is to ensure that the possibility of pipe break within this area is extremely, extremely low such that breaks and cracks can be excluded from the design basis of those portion of piping.

For the APR1400, the applicant's will extend this break exclusion zone for the main steam and feedwater lines to the beyond the upper isolation valve to the auxiliary building wall that they call it a main steam loft house. There's anchoring there.

So, the staff request applicant to justify the variation.

I want to make a note that for the design provision that I earlier mentioned, the DCD's design provision are consistent with the staff guidelines.

However, because the area they expanded, so specifically, in the Request for Additional Information, the staff like to know how those design provision are considered and applied to the results of their PRHA analysis.

And, that particular area, the staff will review that with the PRHA report that when they finally submit because which will provide more detail information of the piping analysis results and that still is an open item.

And, next slide, please?

And, about the technical report methodology for addressing the four potential non-conservatisms, the staff found out the applicant's methodology in addressing the jet plume expansion and distribution of the pressure within the jet plume, the staff found acceptable.

However, the evaluation of blast wave and potential feedback amplification and resonance effects are still open.

And, the status of those open issues, as I mentioned earlier, the break exclusion area as well as the PRHA report, the staff will review them together because they are inter-related and the applicant indicates that it will be submitted probably in the July time frame of this year.

And, as far as the blast wave and potential feedback amplification and resonance effects, this -- for this, the applicant has indicated the resulting dynamic effects using the current technical report methodology would result in a dynamic allowed that's too conservative.

So, they indicate that they will revise the approach and submit a technical report revision, again, probably in the same time frame as the PRHA report.

And this concludes my presentation.

Thank you.

MR. REICHELT: Good afternoon, my name is Eric Reichelt. I'm a Senior Materials Engineer in the Materials and Chemical Engineer Branch of the EIA and NRO.

I am the reviewer for SER Section 3.6.3, leak before break for the APR1400 DCD review.

I would also like to introduce Jay Wallace who's out in the audience from the Office of Research who has provided us with technical assistance with the confirmatory analysis of the applicant's piping evaluation diagrams.

Next slide, please?

The staff reviewed the applicable APR1400 DCD sections and Subsection 3.6.3 to the acceptance criteria and SRP 3.6.3.

The staff reviewed the DCD references for applicability and use.

The staff held public meetings with KHNP and KEPCO about technical issues early on in the review and potential RAIs leading to proposed DCD markups.

The staff found these DCD sections mostly acceptable, most of the technical issues and response to the RAIs and confirmatory items by KHNP were acceptable and were, therefore, closed.

One RAI remains as an open item.

Next slide, please?

The staff questioned the PICEP input file for the surge line fluid temperature against what was provided in the DCD. In addition, the PICEP source code was requested, if available.

This one RAI remains to us as an open item. KHNP will be providing us with their response in the very near future.

The confirmatory analysis will continue upon receiving this response to the open item.

This concludes my presentation. Thank you and I will turn it over to -- my seat over to Tom Scarbrough.

CHAIRMAN BALLINGER: This PICEP code is an EPRI code?

MR. REICHELT: I know the PICEP code --

CHAIRMAN BALLINGER: So, are you likely to get the source code is what I'm asking?

MR. REICHELT: The PICEP code has been utilized in previous designs.

CHAIRMAN BALLINGER: For the past 25 years, but --

MR. REICHELT: Yes.

CHAIRMAN BALLINGER: But --

MR. REICHELT: Yes, Jay, do you happen to --

MR. WALLACE: This is Jay Wallace, Office of Research.

Yes, the PICEP code is an EPRI product. One of the issues that we're having is that the code that we're presently using for doing leak rate analysis, leak bore disagrees slightly with the PICEP code.

The other problem is the PICEP code is quite old and doesn't run on Windows 7 machines.

CHAIRMAN BALLINGER: DOS.

So, what's the resolution?

MR. WALLACE: The resolution there was to fire up the virtual machine under Windows 7 and run the PICEP code for comparison.

CHAIRMAN BALLINGER: Thank you.

MEMBER RICCARDELLA: Excuse me, before you leave --

MR. REICHELT: Oh, sorry.

MEMBER RICCARDELLA: I asked a question earlier about roughly what percentage of lines inside containment are addressed by leak before break versus by normal --

MR. REICHELT: And, it's funny, I had a feeling that I wasn't going to get away from this table without you asking me a question here.

So, the percentage --

MEMBER RICCARDELLA: Just roughly.

MR. REICHELT: We know that, you know, obviously, they're using -- they're looking to apply LBB on the reactor coolant loop, the surge line, the direct vessel injection lines and the shutdown cooling lines.

Their approach is no different than the previous new reactor designs and what's currently been utilized for leak before break out in the operating fleet.

However, as a percentage, I'm sorry, I don't have that answer.

MEMBER RICCARDELLA: As I recall, I think the tendency is to apply it to the larger diameter lines.

MR. REICHELT: Yes.

MEMBER RICCARDELLA: And, not the smaller ones.

MR. REICHELT: Correct.

MEMBER RICCARDELLA: Because it's more difficult to demonstrate --

MR. REICHELT: Right.

MEMBER RICCARDELLA: -- on the smaller diameter ones.

MR. REICHELT: Correct.

MEMBER RICCARDELLA: Okay, thank you.

MR. REICHELT: Okay.

MR. HUR: Excuse me, this is Seokhwan Hur from KEPCO E&C.

I looked at it just a little more explanation for the presentation of the LBB piping within the question scope.

Actually, there are two, I'm not sure of the exact number of the correct piping, there are two uncertain question piping -- current piping from the breaks or from the final rule. So, it becomes a total number to the portion of the piping that cold break and hotleg and the prime rule and associated piping, and there are four piping that is collected to the reactor vessel from the SIT and from the containment break region.

And, there are two shutdown cooling piping and so those are the -- all the LBB piping and there are some more piping like the four piping and pressurize spray piping. But, those are not LBB scope because very small compared to the LBB piping.

So, it's hard to say the percentage of the piping but those are the --

MEMBER RICCARDELLA: Essentially all larger diameter pipes that are LBB?

MR. HUR: Yes, that's correct.

MEMBER RICCARDELLA: Okay, that's what I thought. Thank you.

MR. SCARBROUGH: Good afternoon. I'm Tom Scarbrough. I'm going to go over three topics, specific subsections of Chapter 3.

The first one is the special topics for mechanical components. The staff reviewed the design transients and found that the APR1400 transient occurrences are conservatively designed and, based on the certified System 80+ design transients.

Regarding the computer programs, the staff audited the verification and validation documents for those programs. The staff reviewed sample calculations that are used to benchmark those calculations and prepare audit reports.

There was one specific computer program identified as DPVIB that was used to calculate fluctuating pressure distribution in the downcomer region caused by reactor coolant pump pressure pulsation.

There was initially no benchmarks that were identified for that program during the initial audit and during the follow up audit, there were two separate benchmarks provided for that.

The staff found that that output was in

agreement with the test data and the description of this program has been added to DCD Tier 2 Revision 1, I checked that and it is in there.

Regarding faulted conditions, the staff evaluated the faulted conditions and found that they were in conformance with the boiler and pressure vessel code Section III Appendix F which is the guidance for service loadings for Level B calculations.

MEMBER SKILLMAN: Tom, before you change, on that slide, the first bullet, design transients and the, if you will, the staff's comfort with what is presented for the design certification document, Dr. Ballinger pointed out a couple hours ago that there is this new note at the dead end of Table 3.9-1, page 7 of 7. It's on the Design Cert 3.9-119, is the page location.

And, this is Rev 1, the most recent.

And, that note is, although APR1400 will be operated as a base load plant, the effects of daily load follow operation are accounted for in the structural design and the analysis of the ASME Code Class 1 components, reactor internals and component supports.

My question is this, the overarching

design of this plant is a base load plant. This is a new note and there is no comment in the safety evaluation that opines on, digests, suggests any knowledge of this new note.

So, my question is, what's going on here? Is this -- you might have heard me say an hour or two ago, I find nothing inappropriate with designing this, that could be margin for a plant that would be licenses outside of the United States. But, it just strikes me as that's new and there was no flag in the safety evaluation.

MR. SCARBROUGH: Right.

MEMBER SKILLMAN: Hence, I'm asking the question, how come?

MR. SCARBROUGH: Yes, Tuan Le was the reviewer for this particular section. He's right there at the microphone and, Tuan, have you seen this note before?

MR. LE: That's not a new note. I haven't seen it before. So, is that in the recent revision of the --

MEMBER SKILLMAN: Yes, yes it's Revision 1.

MR. LE: The staff has to review a new addition to confirm those items but have not come across that note yet.

Now, you mention it to look at that now.

MEMBER SKILLMAN: Okay. Well, here's --

MR. LE: And into the validation process.

MEMBER SKILLMAN: Thank you.

Here's why I'm raising the question. You said, Tom, that you found that these transients are conservative and you've based your assessment based on the APR1400 looking at Palo Verde.

MR. SCARBROUGH: Right.

MEMBER SKILLMAN: So, what's different with the plant that would be base loaded versus a plant that say has a 1.5 million cycles over the course of 60 years for small load changes?

MR. SCARBROUGH: Yes, we would have to take a look and see if that would change a number of transients that we've evaluated. Right? If it has a different -- if it has a base load or if it's a following load, you know, is there going to be a difference in the number of transients, we should take a look at that.

So, we'll go back and take a look at that note and talk to the applicant about it and see if that changes any of the determination of the number of occurrences and make sure it's still consistent with the System 80+ which is what was sort of the baseline for this review.

MEMBER SKILLMAN: Oh, it has to do with consistency with Palo Verde but also is the cumulative usage factor calculated as it should be given the idea that this could be a load following plant.

MR. SCARBROUGH: Right.

MEMBER SKILLMAN: So --

MR. SCARBROUGH: Exactly.

MEMBER SKILLMAN: I would like to get that question on the record, please.

MR. SCARBROUGH: Yes, we'll go back and look at that and when we come back again, we'll have an answer for you. We may have to have another RAI or something on this to look at this question and resolve it for you.

MEMBER SKILLMAN: Thank you.

MR. SCARBROUGH: Thank you for identifying that.

MR. SISK: Excuse me, this is Rob Sisk, Westinghouse.

I do want to just kind of clarify on this particular note that the note was added in recognition of previous concerns during the ACRS about clarifying as load follow or base.

So, there are places that you've talked before where we have done some work that follows load following. But, we did not want to have any confusion that because we did that in this case, that we were assuming load follow in the U.S.

We wanted to be clear that there is -- we had done this analysis and put a note in there to distinguish between base load from the U.S. and additional calculations that were being performed.

MEMBER RICCARDELLA: Am I correct in saying, I don't think I saw any difference between the number -- the transients or the numbers of transients in the two Revs. It's just that we had discussions at a prior meeting about base load versus load follow and you put the note in but you didn't really change the transients. So, therefore, the fatigue usage factors wouldn't change, right?

MR. SISK: Yes, correct, not changed.

MEMBER RICCARDELLA: But, you know, I think there's other aspects of load following like concerns with fuel and things that aren't covered in Section 3 design that would need to be addressed before you could load follow?

MR. SISK: Yes, sir.

MEMBER SKILLMAN: Thank you, Pete. Thank

you, Tom.

MR. SCARBROUGH: Okay, thank you.

MEMBER SKILLMAN: Thank you.

MR. SCARBROUGH: We'll follow up on that.

MEMBER SKILLMAN: Okay.

MR. SCARBROUGH: Regarding the next topic is Section 3.9.3 having to do with components, Class 1, 2 and 3 components with the code.

The staff found that the load combinations of the ASME Code Class 1, 2, and 3 components and their supports conformed to the ASME Boiler Code Section III.

We specifically looked at the dynamic system loadings and the dynamic fluid loading. The dynamic system loadings are used in Level D. They are associated with pipe breaks and relief valve actuations.

And, dynamic fluid loadings are associated with more normal valve actuations.

And they are specified and conformed to the boiler code in Section III.

The component supports were found to be designed in accordance with Subsection NF for the boiler code which is the supports subsection.

Now, we did conduct an audit, actually a couple of audits, an initial audit and then a follow up audit recently of the component design specifications in accordance with 10 CFR 52.47 to establish that the design criteria, analytical methods and functional capability satisfied the ASME code requirements and to confirm that the design information from DCD is probably translated into the design specifications.

The initial audit was conducted in 2015 and the follow up audit was conducted relatively recently in 2016 and we're working on the close out audit report for that now.

There are going to be some design specification changes and some DCD updates based on that audit. And, we've prepared an RAI to track those changes so that KHNP will notify us when those design specs are updated and the DCD changes are ready so that we can verify that those changes are incorporated into those specs and DCD.

And so, that will be an open item until we resolve that RAI.

So, that was 3.9.3, now 3.9.6 is the functional design and qualification and IC program.

We evaluated the Section 3.9.6 using the SRP section. The staff found that the DCD provisions for functional qualification of pumps, valves and snubbers specifying the ASME standard QME-1 2007 as endorsed in Reg Guide 1.100 Revision 3 is acceptable.

As I mentioned, we conducted an audit that included the requirements for components to ensure that QME standard is referenced. A number of other areas that we made sure that actuator sizing and things of that nature for valves is properly included. And that report is in preparation.

The DCD description is being updated to -- for the IST program based on the OM code, the 2004 Edition and 2006 Addenda as incorporated in 50.55(a) and it's acceptable for reference in a COL application.

Revision 1 of the DCD includes those updates to describe the program consistent with OM code. There is MOV sizing discussion. There is power operator valve lessons learned discussion in there and also the IST table is being updated.

So, those are -- I've seen those already in the DCD Revision 1 and will confirm those and then close that out.

We had three open items in this section that we sent to you. One was the design spec follow up audit. All those functional design and qualification issues have been addressed and the follow up audit is being prepared and the DCD changes and also the design spec changes will be incorporated.

We wanted to make sure there was references to the QME-1 flow testing and actuators sites and all those sorts of things.

Another open item had to do with the IST table itself. Our original review found that it was not consistent with the OM code and we've -- the latest version they sent in a proposed change back in August of last year. That's consistent with the OM code and we'll be looking over that IST table in Revision 1 to make sure all those changes were made. But, that should be closed out as well.

Our last open item had to do with the ITAAC for pumps and valves. Originally KHNP indicated that they were going to follow the standardized ITAAC the staff was preparing.

Recently, they decided that they would keep the ITAAC they have. So, as a result, we have a few RAIs related to those ITAAC to make sure that, for example, the prototype testing was conducted appropriately and that sort of thing for those ITAAC.

And so, we've sent those RAIs out and KHNP is preparing those responses. So, we imagine that'll be resolved sometime as soon as they get their responses back in to us.

Basically, that's my presentation and I'll turn it over to Tomeka.

MR. CINTRON-RIVERA: Good afternoon. My name is Jorge Cintron. I'm an electrical engineer from the Office of Nuclear Reactor Regulations. And I will be discussing the staff review on Section 3.11.

Section 3.11 provides the APR1400 approach for environmental qualification of mechanical and electrical equipment.

10 CFR 50.49 requires to establish the environmental qualification program to verify the electrical and mechanical equipment important to safety is capable of performing design safety functions on their own environmental conditions.

Regulatory Guide 1.89 provides the guidance for environmental qualification of electrical and mechanical equipment.

In addition, the staff also reviewed nonmetallic parts of safety related mechanical equipment to verify that nonmetallic parts are qualified by testing per ASME standard QME-1 2007.

The staff performed the review following the guidance of SRP 3.11 and the staff officer review and the methodology proposed by the applicant to perform the environmental qualification of mechanical and electrical equipment.

The equipment includes safety-related equipment, non-safety-related equipment as failure on their postulated environmental conditions to prevent satisfactory accomplishments of specified safety functions and certain post-accident monitoring equipment.

Next slide?

The staff is currently reviewing an open item with regard to Section 3.11. Regulatory Guide 1.89 provides the acceptable methods for environmental qualifications. The guidance endorsed IEEE 323 1974 as an acceptable method for environmental qualification of electrical equipment.

The applicant deviates from the Regulatory Guide and proposed the use of IEEE 323 2003.

The staff performed an assessment of both the standards to identify technical differences and are confident that IEEE 323 2003 and the 1974 and issued a follow up RAI requesting the applicant to provide justification of the technical differences between the 1974 and 2003 version.

Next slide?

The staff is --

MEMBER STETKAR: Jorge, just out of curiosity, why hasn't the staff updated the Reg Guide since it's based on information that's now 43 years old?

MR. CINTRON-RIVERA: Good question.

MEMBER STETKAR: Probably older than you, not older than me.

MR. CINTRON-RIVERA: The 2003 version was never endorsed by the staff. However, there is a 2016 that is being issued, and the staff will consider endorsing the new guidance.

MEMBER STETKAR: Okay.

MR. CINTRON-RIVERA: I guess for the purpose of 2003 version versus the one that version now the staff were more involved in the development of the 2016 version. So, we believe that it will be considered for endorsement.

MEMBER STETKAR: Thank you.

MR. CINTRON-RIVERA: So, right now, the staff is currently evaluating the applicant's revised response to develop RAI which is an open for Section 3.11.

With that, it concludes my presentation.

MR. STUTZCAGE: Hi, I'm Ed Stutzcage and

I'm the Radiological reviewer for Section 3.11.

Next slide, please?

Staff reviewed the applicant's methodology and results for calculating the total integrated dose to equipment within the EQ program.

For the normal operation dose, the applicant calculated the normal operation doses at rooms based on radiation sources in the plant, considering gamma and neutron radiation.

Normal operation doses for equipment was calculated assuming the highest dose within the room using a similar methodology to that use to determine Chapter 12 radiation zone maps, except that, for the EQ analysis, 1 percent failed fuel is assumed consistent with Reg Guide 1.89.

Staff reviewed the normal operation sources and dose values and found them to be acceptable with the exception of the open items.

For the accident doses, they're based on the most limiting design basis accident for each radiated plant, for most areas, that the LOCA.

Auxiliary building sources include post-accident fluid recirculating the shutdown cooling system, safety injection system and containment spray system.

The staff verified these sources were consistent with the guidance in Reg Guide 1.183.

The total integrated dose is the sum of the 60-year normal operation dose plus the limiting accident doses.

Next slide, please?

For most areas of the plant, staff finds the applicant's normal operation dose values for equipment qualification to be acceptable.

Outstanding issues include inconsistencies with Chapter 12 information regarding normal operation neutron doses under the refueling floor.

As part of our Chapter 12 review, we did review the calculation package that indicated there'd be neutron streaming past the shield blocks into the -- on to the operating floor and the EQ analysis didn't seem to include that neutron dose. So, the applicant's looking into that.

Another issue is doses within the auxiliary building. Their accidents didn't appear to consider radiation streaming through containment penetrations.

But, we found that doing some Microshield calculations and some other just reviewing their TID values.

So, we asked the applicant to resolve that issue.

We requested the applicant provide additional information regarding how some of the post-accident gamma dose rate information was determined.

And, there was unjustified assumptions for post-accident fluid leakage rate outside of containment. Their post-accident fluid leakage assumption was different in Section 3.11 than it was in the Chapter 15 analysis. So, we asked the applicant to explain the difference and justify it.

That concludes my presentation.

MEMBER MARCH-LEUBA: If calculating the neutron streaming through holes and deep shielding is not in the calculation, have you considered asking the applicant for experimental data on their existing reactors?

MR. STUTZCAGE: Didn't ask them for experimental data.

MEMBER MARCH-LEUBA: I mean, it's a very easy thing to how a reactor is working or measured because those calculations aren't difficult.

MR. STUTZCAGE: Right, right, I agree.

We can look into that. I can look into that.

MEMBER MARCH-LEUBA: And, probably be better for them the calculation actually because things tend to smear.

MR. STUTZCAGE: Yes, right, right, agree. Thanks.

MS. TERRY: That concludes our presentation. Unless you have anything else you would like to discuss with us.

CHAIRMAN BALLINGER: Yes, we've finished very, very early and myself as well as other members have looked at the SER pretty carefully.

But, and I'm sure we would have raised questions if we had seen something in there that we thought was important enough for you to discuss but you hadn't discussed.

But, for the record, I'd like to ask you whether or not, based on your presentations and the ones that haven't been given, are there any other potential -- well, not any other -- any potential important issues that were not discussed today?

MR. SCARBROUGH: This is Tom Scarbrough.

I don't know of any issues that we have concerns with. If we did, we would definitely raise them with you.

MS. TERRY: Yes, most of the other

sections didn't have any -- no specific issues.

CHAIRMAN BALLINGER: Yes, I didn't see any, but then again, who knows.

So, I guess we should get the bridge line open.

MR. BROWN: Bridge open.

CHAIRMAN BALLINGER: Good. While we're doing that, is there anybody in the room that would like to make a comment? I have to face the mic.

(No response.)

CHAIRMAN BALLINGER: Hearing none, the bridge line is open. Is there anybody out on the bridge line that would like to make a comment?

(No response.)

CHAIRMAN BALLINGER: You're sure it's open, no crackling or anything?

MR. BROWN: Open.

CHAIRMAN BALLINGER: Okay, good. Well, I heard a crackle, good.

Hearing none, I'd like to go around the table for any other member comments.

Joy?

MEMBER REMPE: No comments.

MEMBER MARCH-LEUBA: I have no further comments.

MEMBER STETKAR: Nothing at all, and especially congratulations to the applicant for a heroic effort to go through all of that material in the morning.

MEMBER RICCARDELLA: No comments.

CHAIRMAN BALLINGER: Dana?

MEMBER POWERS: No.

CHAIRMAN BALLINGER: Dick?

MEMBER SKILLMAN: Thank you, no further comment.

CHAIRMAN BALLINGER: And, I'd like to congratulate the staff and the applicant as we've gone through this process chapter by chapter by chapter, you can see an obvious significant improvement in the process itself.

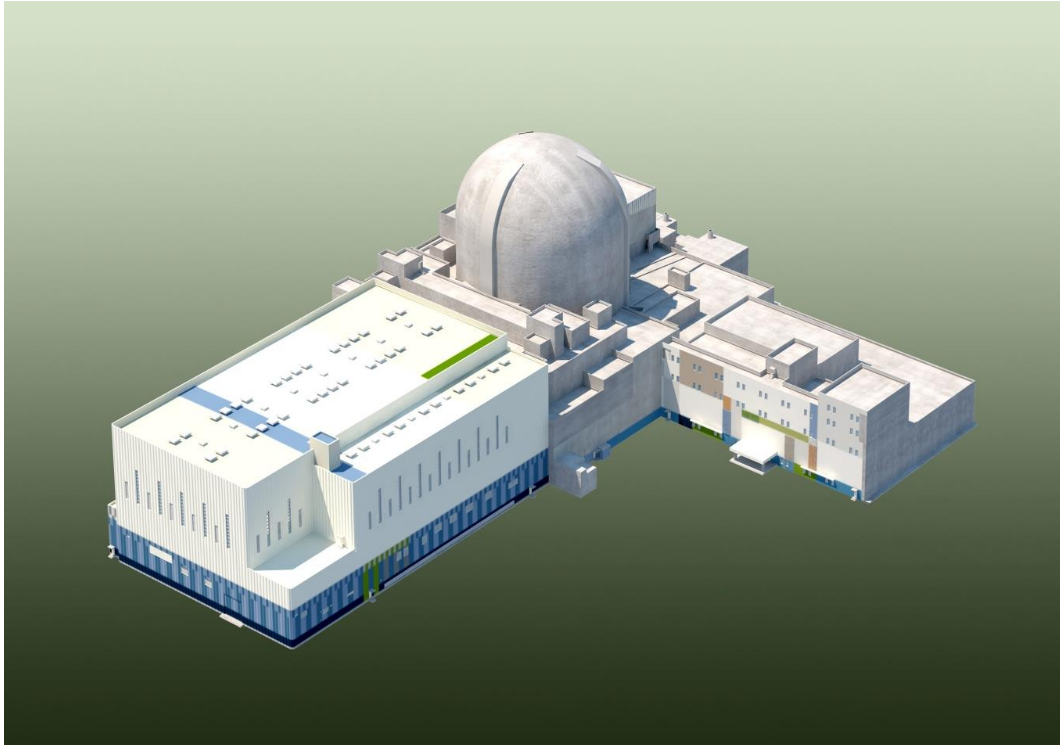
So, I think that's reflected in today's presentations. So congratulations, and I can't turn around and congratulate the applicant.

But, that being the case, we are adjourned.

(Whereupon, the above-entitled matter went off the record at 2:06 p.m.)

NON-PROPRIETARY

APR1400 DCA
Chapter 3: Design of Structures, Systems, Components, and Equipment



KEPCO/KHNP

June 5, 2017

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Overview of Chapter 3

● Section Overview

Section	Title	Major Contents	Presenter
3.1	Conformance with Nuclear Regulatory Commission General Design Criteria	• Conformance with Nuclear Regulatory Commission General Design Criteria 1 through 64	Jinkyoo Yoon
3.2	Classification of Structures, Systems, and Components	• Classification of Structures, Systems, and Components according to nuclear safety classification, quality groups and seismic category	Jinkyoo Yoon
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	Jinkyoo Yoon
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	Jinkyoo Yoon
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	Jinkyoo Yoon

Overview of Chapter 3

● Section Overview

Section	Title	Major Contents	Presenter
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	Jinkyoo Yoon
3.7	Seismic Design	• Seismic input motions • Seismic analysis methodology and results of seismic Category I Structures with generic soil sites • Seismic analysis methodology of seismic Category I subsytems • Seismic monitoring system	Yongsun Lee
3.8	Design of Category I Structures	• Design features of Category I Structures including concrete containment, steel containment, internal structures of containment, other seismic Category I structures, and foundations	Hoonin Cho
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.	Hongsun Park

Overview of Chapter 3

● Section Overview

Section	Title	Major Contents	Presenter
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	Bosung Choi
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.	Bosung Choi
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.	Bosung Choi
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	Hongsun Park

Overview of Chapter 3

● List of Submitted Documents

Document No.	Title	Rev.	Type	ADAMS Accession No.
APR1400-K-X-FS-14002-P&NP	APR1400 Design Control Document Tier 2: Chapter 3	1	DCD	-
APR1400-K-X-IT-14001-P&NP	APR1400 Design Control Document Tier 1, Section 2.2	1	DCD	-
APR1400-E-S-NR-14001-P&NP	Seismic Design Bases	1	TER	ML17094A154
APR1400-E-S-NR-14002-P&NP	Finite Element Seismic Models for SSI Analyses of the NI Buildings	1	TER	ML17094A157
APR1400-E-S-NR-14003-P&NP	SSI Analysis Methodology and Results of NI Buildings	1	TER	ML17094A115
APR1400-E-S-NR-14004-P&NP	Evaluation of Effects of HRHF Response Spectra on SSCs	2	TER	ML17094A116
APR1400-E-S-NR-14005-P&NP	Evaluation of Structure-Soil-Structure Interaction (SSSI) Effects	1	TER	ML17094A117
APR1400-E-S-NR-14006-P&NP	Stability Check for NI Common Basemat	2	TER	ML17094A118
APR1400-Z-M-NR-14009-P&NP	Comprehensive Vibration Assessment Program for the Reactor Vessel Internals	1	TER	ML17094A142
APR1400-E-X-NR-14001-P&NP	Equipment Qualification Program	0	TER	ML13304A908
APR1400-E-N-NR-14003-P&NP	Evaluation Methodology of Jet Impingement Loads on SSCs	0	TER	ML15279A003
APR1400-E-N-NR-14004-P&NP	Summary Report of High-Energy Piping Rupture Analysis	0	TER	ML15243A015
APR1400-Z-M-NR-14016-P&NP	Leak-Before-Break Evaluation for Surge Line	0	TER	ML15009A122

3.1 Conformance with Nuclear Regulatory Commission General Design Criteria

NON-PROPRIETARY

❖ General Description

- This section discussed the extent to which the design criteria for the plant structures, systems, and components important to safety meet the NRC "General Design Criteria for Nuclear Power Plants" specified in Appendix A to 10 CFR Part 50

- For each criterion, a summary was provided to show how the principal design features meet the criterion in the relevant DCD sections

3.2 Classification of Structures, Systems, and Components

The APR1400 structures, systems, and components (SSCs) are classified according to seismic category, quality groups, and nuclear safety classification.

3.2.1 Seismic Classification

Seismic classification is identified to meet GDC 2 in accordance with RG 1.29

3.2.2 System Quality Group Classification

System quality group classification is identified to meet GDC 1 in accordance with RG 1.26

3.2.3 Safety Class

Fluid system components important to safety are classified in accordance with ANSI/ANS-51.1-1983

3.2.4 Classification Listings

Component classifications including quality assurance and codes and standards are provided in DCD Table 3.2-1.

3.3 Wind and Tornado Loadings

3.3.1 Wind Loadings

The design wind loadings on the surfaces of the SSCs subject to wind are determined in accordance with ASCE/SEI 7-05

- **Design Wind Velocity**
  - 50-year 3-second wind gust speed : 64.8 m/sec (145 mph)
  - Wind speed at 10 m (33 ft) above ground

3.3.2 Tornado Loadings

The APR 1400 standard and site-specific plant is designed to protect SSCs from tornadoes and hurricanes

- **Applicable Design Parameters**
  - Design Basis Tornado (NRC RG 1.76)
    - Max. horizontal wind speed for 10 million years : 102.8 m/s (230 mph)
  - Design Basis Hurricane (NRC RG 1.221)
    - Max. wind speed for 10 million years: 116 m/s (260 mph)

3.4 Water Level (Flood) Design

3.4.1 Flood Protection and Evaluation

- **Design Bases**
  - The design basis flood level at the reactor site is determined in accordance with NRC RG 1.59 and ANSI/ANS 2.8.
  - Design Basis Flood Level
    - 0.3 m (1 ft) below the plant grade
  - Maximum groundwater level
    - 0.61 m (2 ft) below the plant grade
- **Flood Protection from External Sources**
  - The flood protection measures are designed in accordance with NRC RG 1.102.

3.4 Water Level (Flood) Design

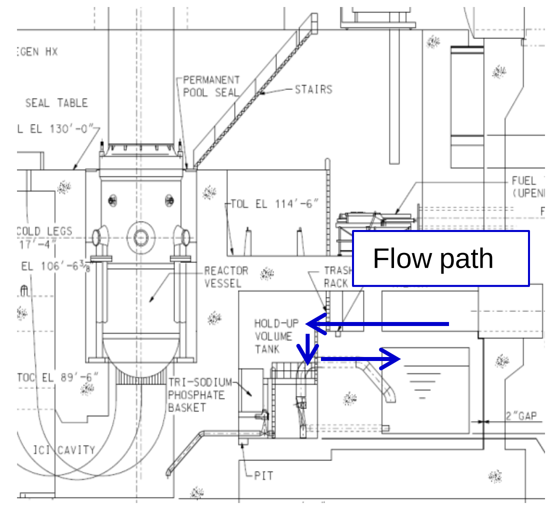
3.4.1 Flood Protection and Evaluation (Cont'd)

- **Flood Protection from Internal Sources**
  - The safety-related SSCs are designed to withstand the effects of flooding due to natural phenomena or onsite equipment failures without loss of the capability to perform their safety-related functions.
  - The flood protection mechanisms from internal sources are designed in APR1400 ;
    - Structural enclosure or barrier walls, drainage system including Emergency Overflow Line(EOL), emergency sump, internal curbs or ramp and watertight doors.
  - Potential flooding sources include flood water due to postulated pipe failure, inadvertent operations of fire protection systems, failure of non-seismic piping.

3.4 Water Level (Flood) Design

3.4.1 Flood Protection and Evaluation (Cont'd)

- **Evaluation of Internal Flooding**
  - Reactor Containment Building(RCB)
    - Flooding source flows to the EL.100 ft of RCB, and to the HVT and IRWST through two 24-inch spillways.
    - Limiting flood source is considered as LOCA.
    - Flood height is determined to be 0.61 m (2 ft) from the EL.100 ft considering the floodable area and LOCA volume.



  - Auxiliary Building(AB)
    - Flooding source flows to the bottom level (EL.55 ft) by drainage system and EOL.
    - Limiting flood source is the water source of the IRWST.
    - Flood height is determined to be 2.74 m (9 ft) from the EL.55 ft considering total water volume of IRWST and floodable area.
    - Each quadrant of AB at EL.55 ft is separated by division walls and watertight doors are designed to prevent flooding source from spreading to adjacent quadrants.

3.5 Missile Protection

3.5.1 Missile Selection and Description

- **Missile Protection**
  - Applicable code : 10 CFR 50 Appendix A, GDC 2 and 4.
  - Missile protection is provided for SSCs important to safety
    - Potential missiles do not cause the release of significant radioactivity or,
    - Do not prevent the safe shutdown of reactor
  - Missile protections are accomplished as following:
    - Minimizing the missile generation source by equipment design
    - Orienting or physically separating potential missile sources from safety related equipment and components
    - Containing the potential missiles through protective shields or barriers
    - Hardening of safety-related equipment and components to withstand missiles impact

3.5 Missile Protection

3.5.1 Missile Selection and Description(Cont'd)

- **Missile Protection(Cont'd)**
  - Internally Generated Missile
    - Internally generated missiles are categorized as two types missiles, rotating and pressurized components.
    - If the probability of missile generation P_1 is maintained less than 10^{-7} per year, missile is not considered statistically significant.
    - Potential missiles inside containment are summarized in Table 3.5-1.
    - Structures inside the containment, secondary shield wall, refueling pool wall, structural beams, and floor slabs are served as missile shields.

3.5 Missile Protection

3.5.1 Missile Selection and Description(Cont'd)

- **Missile Protection(Cont'd)**
  - Turbine Missiles
    - The turbine generator is placed with favorable orientation so that all essential SSCs are excluded from the low-trajectory turbine missile strike zone as defined by R.G 1.115.
    - Essential SSCs outside the RCB to be protected from externally generated missiles are listed in Table 3.5-4.
  - Missiles Generated by Tornadoes and Extreme Winds
    - Safety-related SSCs are protected against the impact generated by tornado or hurricane missiles.
    - Provides reasonable assurance for the protection according to 10 CFR Part 50, Appendix A GDC 2, 4 and 10 CFR 52.47(b)(1)
    - Selected missile types - Pipe, Automobile, Solid Steel Sphere

3.5 Missile Protection

3.5.1 Missile Selection and Description(Cont'd)

- **Missile Protection(Cont'd)**
  - Site Proximity Missile (Except Aircraft)
    - Evaluation of the potential for site proximity explosions and missiles by the COL Applicant
  - Aircraft Hazards
    - Justification for the site-specific aircraft hazard and an aircraft hazard analysis in accordance with the requirements of NRC RG 1.206 by the COL Applicant

3.5.2 Structure, Systems, and Components to be Protected from Externally Generated Missiles

- **Structures used to protect safety-related SSCs meet the requirements of NRC RGs 1.13, 1.27, 1.115 and 1.117**
- **Essential SSCs protected against missile impact are listed in DCD Table 3.5-4**

3.5 Missile Protection

3.5.3 Barrier Design Procedures

- **Missile barriers are designed with sufficient strength and thickness to prevent local damage including perforation, spalling and scabbing, and overall damage**
- **Evaluation of Local Structural Effects**
  - Concrete Barriers
    - Estimation of the depth of missile penetration
    - Assessment of secondary missile by spalling
  - Steel Barriers
    - Not use in the APR1400 design
- **Overall Damage Prediction**
  - For the evaluation of overall response of reinforced concrete barriers under impact and impulse load, nonlinear and elasto-plastic response of structures is used.

3.6 Protection against Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.1 Plant Design for Protection against Postulated Piping Failures in Fluid Systems Inside and Outside the Containment

- **Design Bases**
  - Codes & Standards:
    - 10 CFR 50, Appendix A, GDC 2, 4
    - SRP 3.6.1, 3.6.2, 3.6.3, BTP 3-3 and 3-4
    - ANSI/ANS 58.2 -1988
  - High and moderate energy fluid systems are summarized in Table 3.6-1.

- **Protection of Essential SSCs**
  - The design to protect essential SSCs from the effect of postulated break is basically achieved by separation, physical barriers, or pipe whip restraints(PWR).

3.6 Protection against Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

- **Postulated pipe break locations are determined in accordance with BTP 3-4, Part B**
  - Terminal ends
  - Intermediate location based on stress analysis results
- **The criteria used to define locations and configurations of pipe rupture are provided in Subsection 3.6.2.1**
- **The analytical methods to define forcing functions are described in Subsection 3.6.2.3**

NON-PROPRIETARY

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping (Cont'd)

- **The MSVH is the only area that meets the break exclusion criteria described in BTP 3-4 B.A.(ii) for ASME Class 2 piping**
  - The design of break exclusion is applied for fluid system piping in containment penetration areas, between containment wall and auxiliary building anchor wall beyond the isolation valve.
- **Forcing functions for pipe thrust and jet loads are based on ANSI 58.2 Appendix B**
- **Dynamic and environmental effects due to HELB and MELB are summarized in the pipe rupture analysis report**
  - Dynamic effects analysis; pipe whip, jet impingement and sub-compartment pressurization
  - Environmental effects analysis; flooding, environmental qualification

* MSVH: Main Steam Valve House

3.6 Protection against Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping (Cont'd)

- **Non-conservatism of jet impingement model in ANSI 58.2 will be addressed as an open item**
  - Blast Waves
  - Jet Plume Expansion and Zone of Influence
  - Distribution of Pressure within the Jet Plume
  - Jet Dynamic Loading including Potential Feedback Amplification and Resonance Effects.
- **The current technical report and related RAI responses will be revised to address the above issues**

3.6 Protection against Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping (Cont'd)

- **Pipe Whip Restraints**
  - Provide to protect the safety-related component against the effects of pipe whipping during postulated pipe break
  - Consideration of pipe whip restraints design
    - Pipe break blowdown thrust, functional requirement, deformation limitations
    - Properties of whipping pipe, capacity of the support structure
  - Allowable stresses
    - Limitation of strain of energy-absorbing members
    - Application of AISC N690 for non-energy-absorbing members

3.6 Protection against Dynamic Effects Associated with the Postulated Rupture of Piping

NON-PROPRIETARY

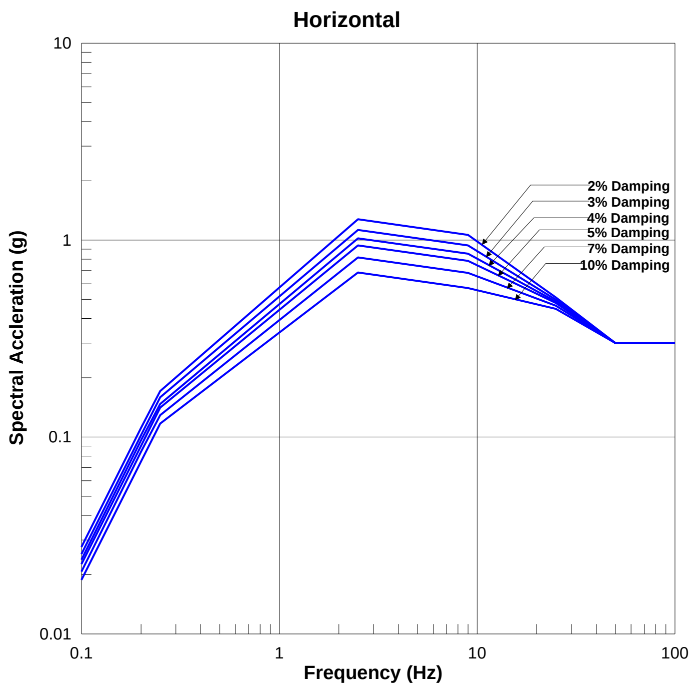
3.6.3 Leak-Before-Break Evaluation Procedure

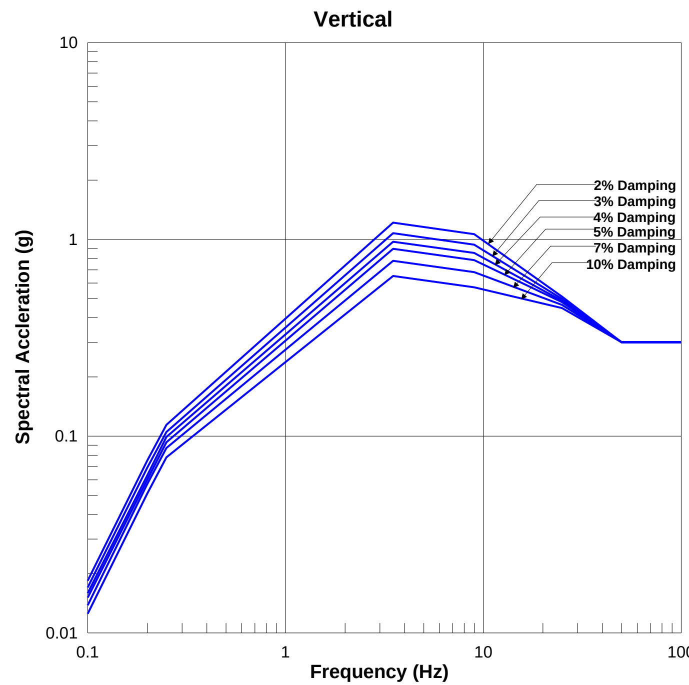
- **LBB analysis is used to eliminate the dynamic effects of pipe breaks**
- **This subsection describes how the piping system meets the LBB criteria in accordance with SRP 3.6.3 and demonstrates that the probability of pipe rupture is extremely low**
- **LBB is applied to the following piping systems:**
  - Reactor coolant loop piping
  - Surge line
  - Direct vessel injection line inside containment
  - Shutdown cooling line inside containment
- **The method of PEDs (Piping Evaluation Diagrams) allows for the evaluation of the piping system incorporating LBB considerations into the piping design**
- **The LBB evaluation is consistent with the requirements of SRP 3.6.3 and NUREG-1061, Volume 3**

3.7 Seismic Design

3.7.1 Seismic Design Parameters

- **Certified Seismic Design Response Spectra (CSDRS)**
  - Peak ground acceleration (PGA) : 0.3g
  - Zero period acceleration frequency : 50 Hz
  - CSDRS based on the NRC RG 1.60 with enrichment in the high frequency range.

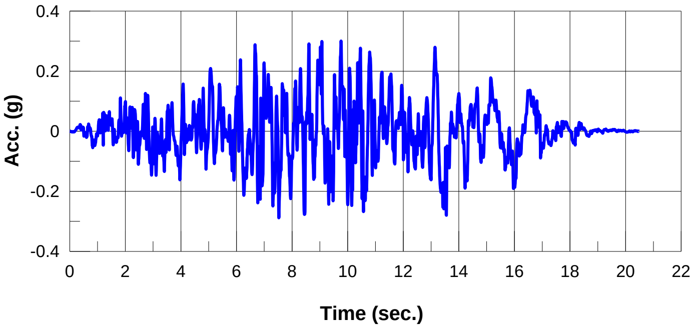


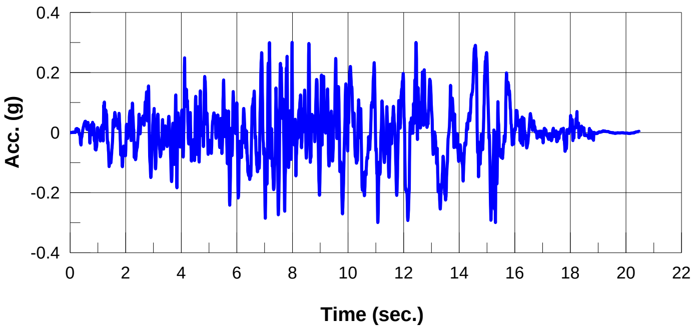


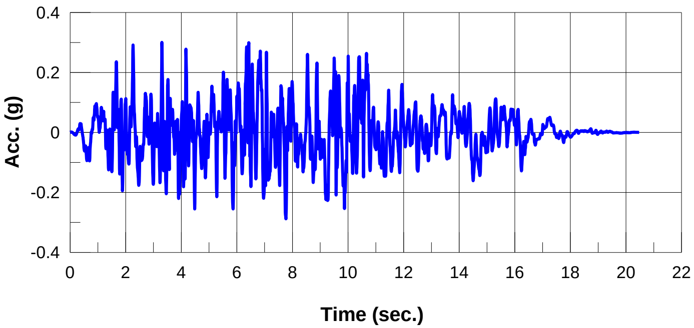
3.7 Seismic Design

3.7.1 Seismic Design Parameters

- **CSDRS Time Histories**
  - Generation criterion : Single set of time histories (Option 1, Approach 1) in NRC SRP 3.7.1
  - Initial seed motions : Northridge earthquake time histories
  - Cross-correlation coefficients :
    - EW-NS : 0.032 < 0.16
    - NS-VT : 0.029 < 0.16
    - EW-VT : 0.079 < 0.16



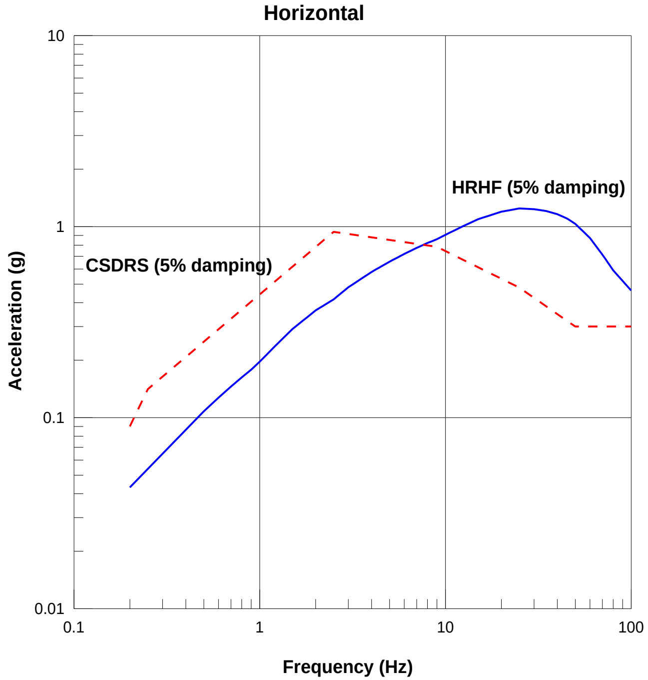


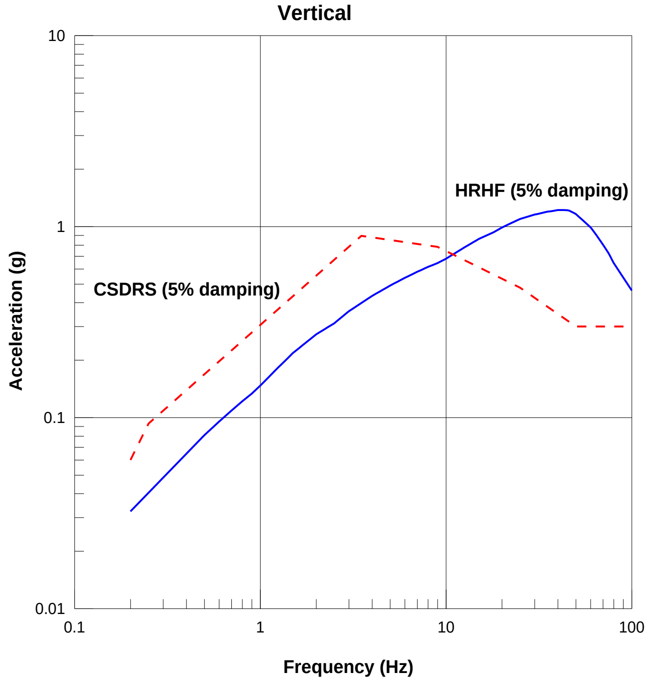


3.7 Seismic Design

3.7.1 Seismic Design Parameters

- **Hard Rock High Frequency (HRHF) Response Spectra**
  - GMRS for some Central and Eastern United States rock sites show higher amplitude at high frequency than the CSDRS.
  - Peak ground acceleration : 0.46g

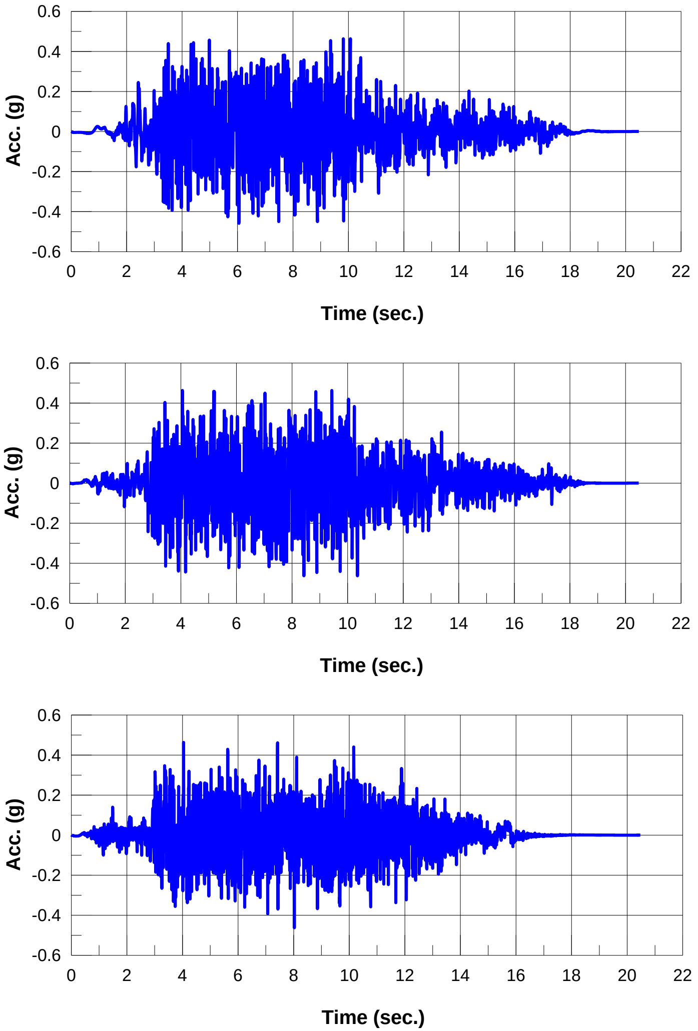




3.7 Seismic Design

3.7.1 Seismic Design Parameters

- **HRHF Time Histories**
  - Generation criterion : Single set of time histories (Option 1, Approach 1) in NRC SRP 3.7.1
  - Initial seed motions : Nahanni earthquake time histories
  - Cross-correlation coefficients :
    - EW-NS : 0.028 < 0.16
    - NS-VT : 0.031 < 0.16
    - EW-VT : 0.036 < 0.16



3.7 Seismic Design

3.7.1 Seismic Design Parameters

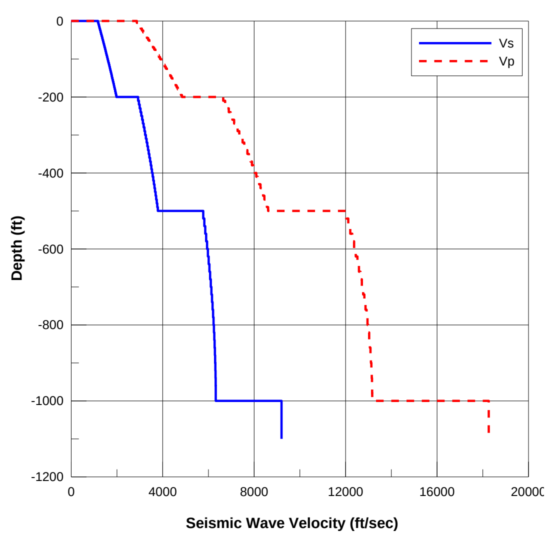
- **Critical Damping Values**
  - Damping values for structural material based on NRC RG 1.61.
  - Soil damping values which are used in soil-structure interaction analysis recommended by EPRI TR-102293.

Structural material	SSE	OBE
Welded steel or bolted steel with friction connections	4 %	3 %
Bolted steel with bearing connections	7 %	5 %
Prestressed concrete	5 %	3 %
Reinforced concrete	7 %	4 %

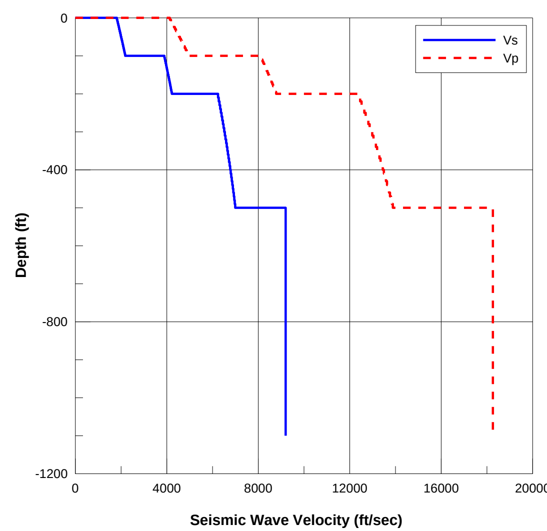
3.7 Seismic Design

3.7.1 Seismic Design Parameters

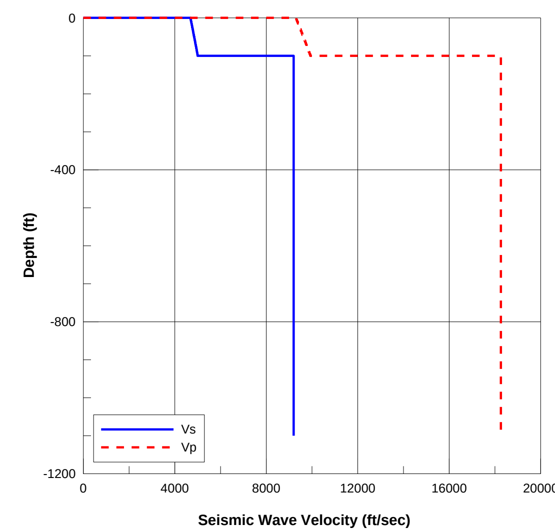
- **Supporting Media for Seismic Category I Structures**
  - Founded directly on rock or competent soil
  - Eight soil profiles and one fixed-base condition



S01 Soil Profile



S04 Soil Profile



S09 Soil Profile

3.7 Seismic Design

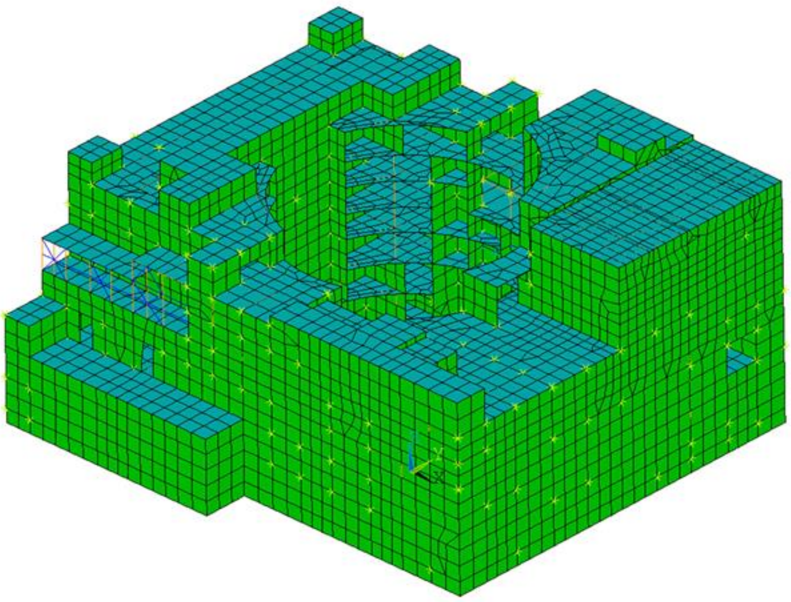
3.7.2 Seismic System Analysis

- **Seismic Analysis Methods**
  - Complex frequency response analysis method : used for three-dimensional soil-structure interaction analyses including a fixed-base analysis to obtain seismic responses of the standard plant seismic Category I structures
- **Analysis Models of Seismic Category I Structures**
  - Safety-related structures model : three-dimensional finite element models (FEMs).

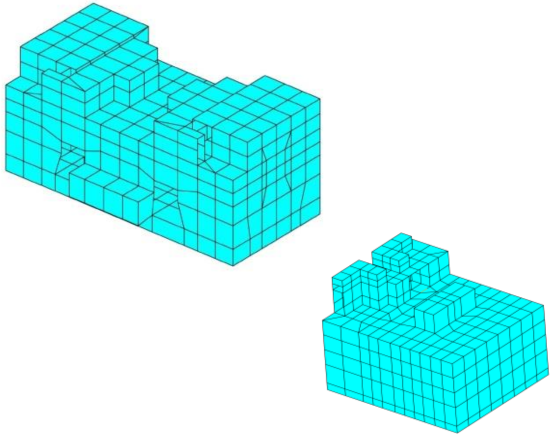
3.7 Seismic Design

3.7.2 Seismic System Analysis

- **Analysis Models of Seismic Category I Structures**



Auxiliary Building



Emergency Diesel Generator Building and Diesel Fuel Oil Storage Tank Room

3.7 Seismic Design

3.7.2 Seismic System Analysis

- **In-structure Response Spectra**
  - Generation method and software : Complex frequency response method and ACS SASSI software at wall and floor locations in the FEMs
  - Effect of potential concrete cracking : uncracked and cracked concrete stiffness cases
  - Design-basis ISRS : Enveloped ISRS of 9 soil cases for both uncracked and cracked concrete stiffness cases

3.7 Seismic Design

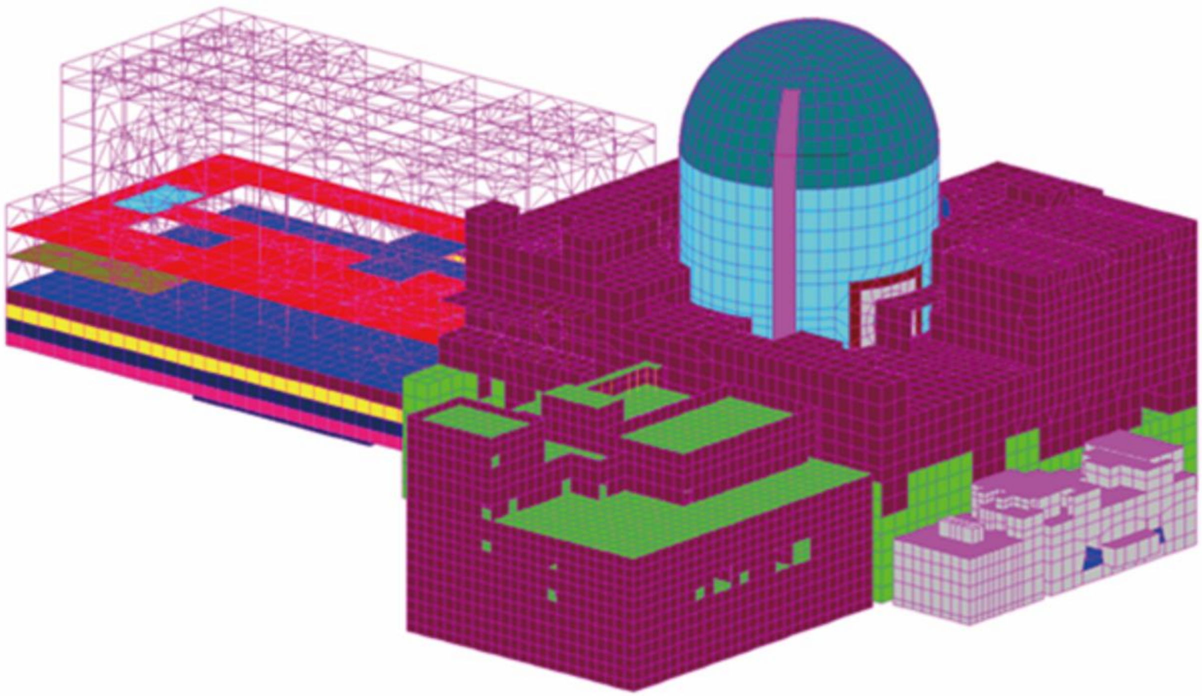
3.7.2 Seismic System Analysis

- **Interaction of Non-Seismic Category I Structures with Seismic Category I Structures**
  - Seismic Category II structures : Turbine generator building and compound building with a 900 mm (3 ft) gap on nuclear island
  - Analysis and design of seismic Category II structure : Prevention their failure under SSE conditions
  - Structure-soil-structure interaction (SSSI) analysis : Evaluation on the nuclear island structures due to presence of adjacent non-seismic Category I structures, the SSSI analysis using the coupled model
  - SSSI effect
    - Increased seismic response of emergency diesel generator building by nuclear island
    - Negligible effect to nuclear island by non-seismic Category I structures

3.7 Seismic Design

3.7.2 Seismic System Analysis

- **SSSI Analysis Model**

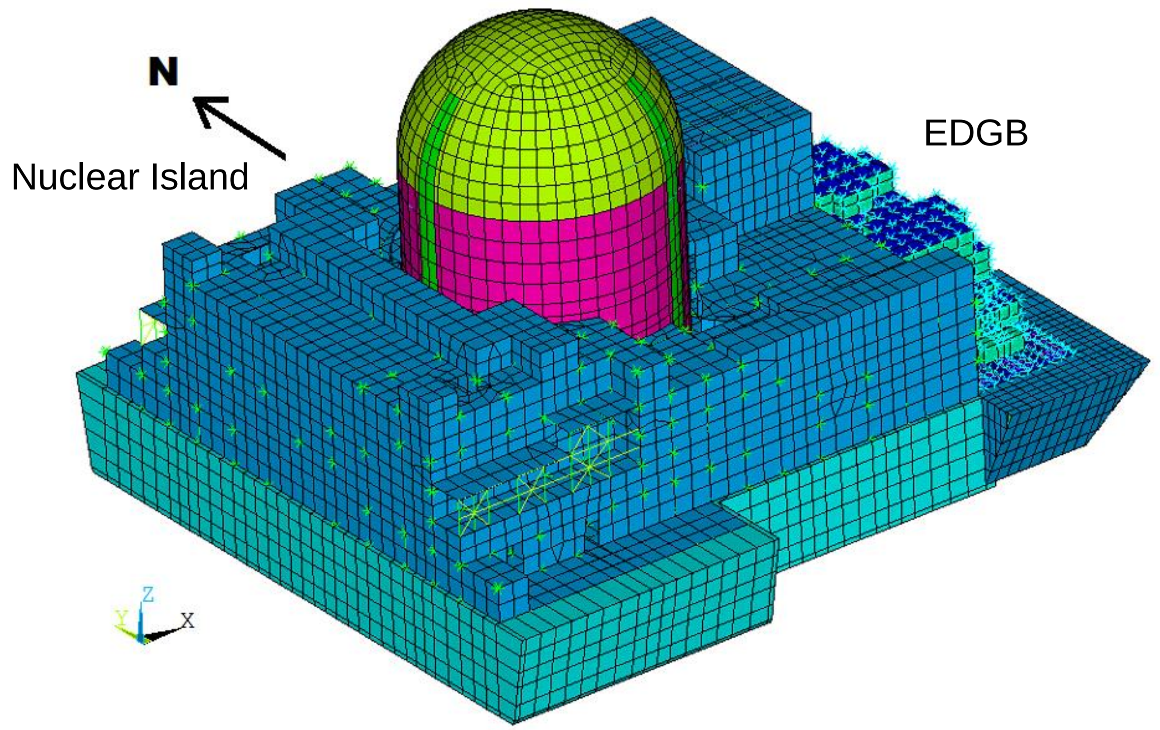


SSSI Analysis Model for Power Block with Surface-supported Foundation Condition

3.7 Seismic Design

3.7.2 Seismic System Analysis

- **SSSI Analysis Model**



SSSI Analysis Model for Nuclear Island and EDGB with Excavated Soil Volume

3.7 Seismic Design

3.7.2 Seismic System Analysis

- **Incoherent SSI Analysis with HRHF Seismic Input Motion**
  - APR1400 HRHF response spectra : 0.8-fractile GMRS based on the 2011 EPRI Report "Evaluation of seismic hazard at Central and Eastern US nuclear power sites"
  - Incoherent input motion and coherency functions : Developed by Abrahamson (2007) using INCOH code, and incoherent ground motion vector input to ACS SASSI software
  - Evaluated structures, systems and components : Nuclear island (RCB, AB, RCS) and EDGB

3.7 Seismic Design

3.7.3 Seismic Subsystem Analysis

- **Seismic Analysis Methods of Seismic Category I Subsystem**
  - Response spectrum analysis
  - Time-history analysis
  - Equivalent static analysis method

3.7 Seismic Design

3.7.4 Seismic Instrumentation

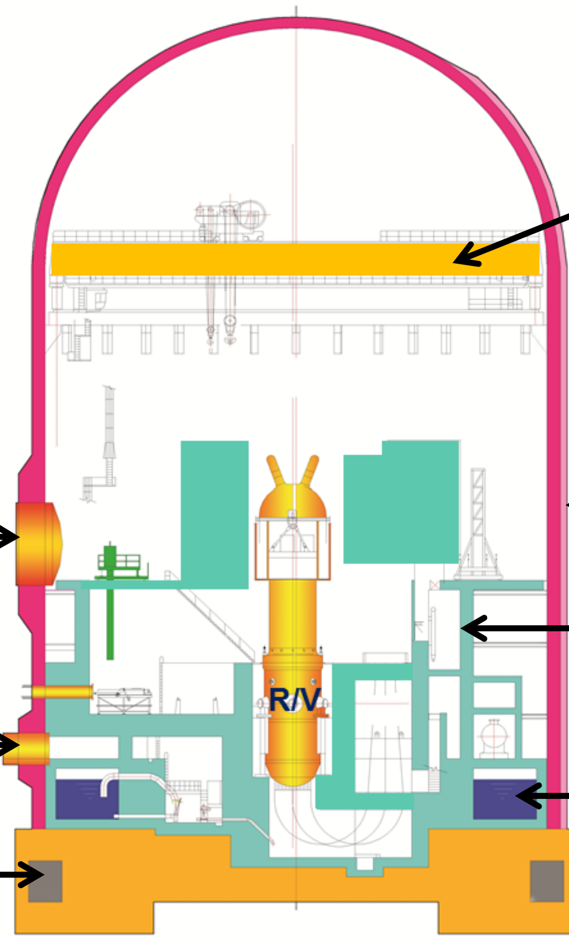
- **Seismic Monitoring System**
  - Seismic monitoring system design in accordance with NRC RG 1.12.
  - Time-history accelerographs installed at appropriate locations
    - Free-field
    - Containment structure
    - Other seismic Category I structures

3.8 Design of Category I Structures

3.8.1 Concrete Containment

- **Structure Description**

Polar Crane & RV



3.8 Design of Category I Structures

3.8.1 Concrete Containment

- **Containment Dimensions**
  - Inside diameter of containment: 45.72 m (150 ft)
  - Inside height of containment: 76.66 m (251.5 ft) from the top of base slab to the ceiling of dome apex
  - Thickness of containment wall: 1.37 m (4 ft 6 in)
  - Dome thickness: 1.22 m (4 ft)

- **Structural Analysis**
  - The 3D Finite Element models (ANSYS) are developed to perform the structural analysis of the containment structure.

3.8 Design of Category I Structures

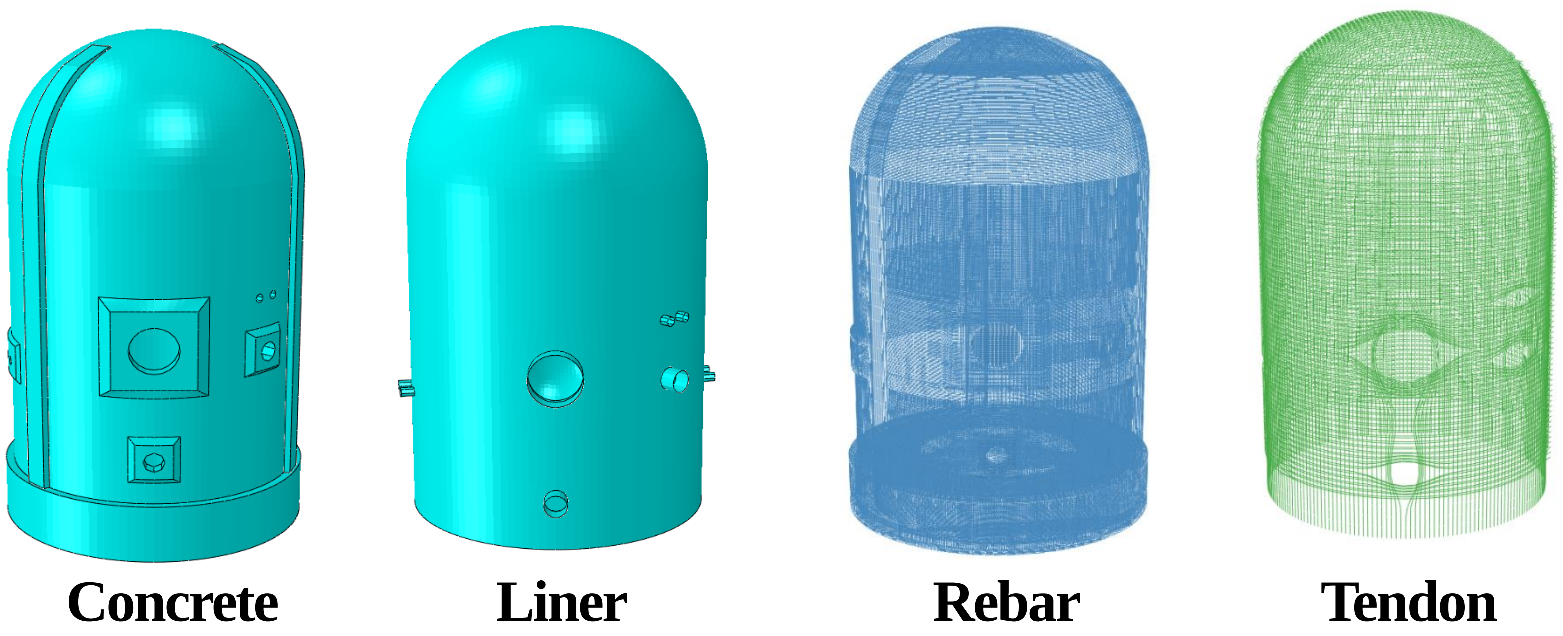
3.8.1 Concrete Containment

- **Combustible Gas Control Inside Containment**
  - Structural Integrity of the containment structure (10CFR50.44, RG 1.216)
    - Factored Load Category requirements of ASME Code Section III, Division 2 Subarticle CC-3720 (for concrete containment)
  - Acceptance Criteria (Leak-tightness criteria)
    - Allowable strain of liner plate: Subarticle CC-3720 of ASME Code
  - Temperature-Dependent Material Property Degradation (NUREC/CR-6906)
    - Maximum temperature = 350 °F

3.8 Design of Category I Structures

3.8.1 Concrete Containment

- **Combustible Gas Control Inside Containment (ABAQUS)**



3.8 Design of Category I Structures

3.8.1 Concrete Containment

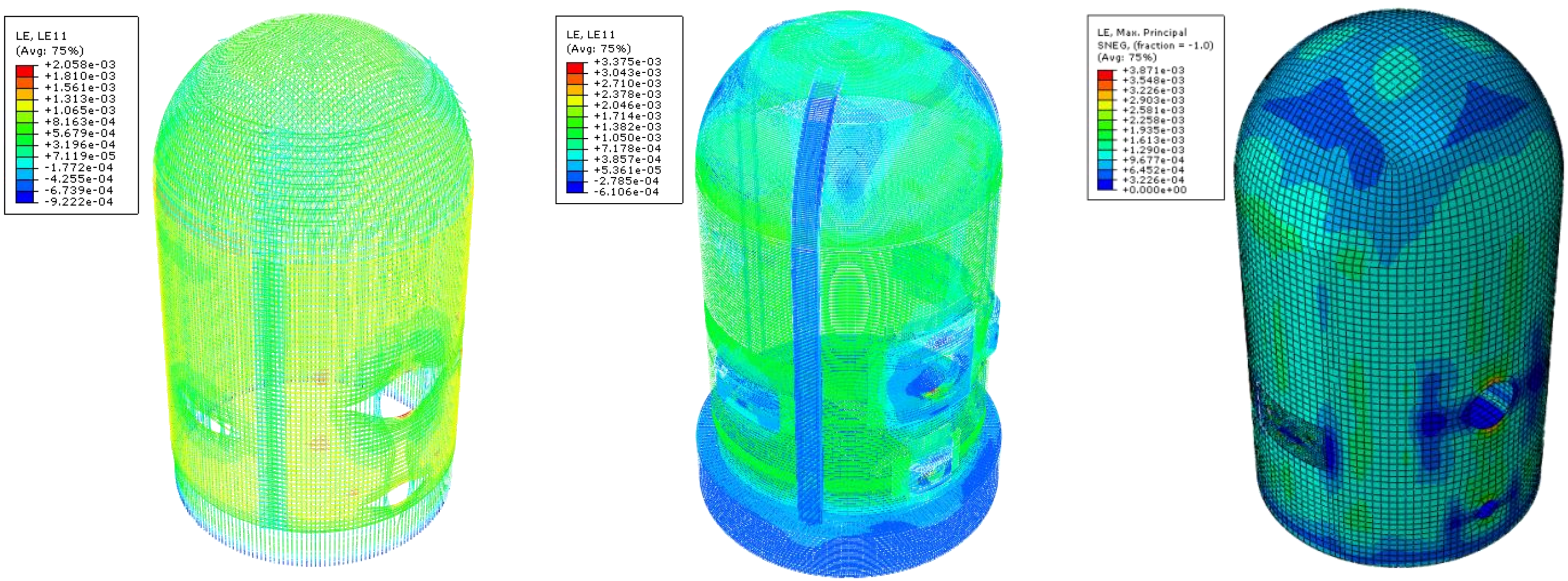
- **Combustible Gas Control Inside Containment**
  - Max Pressure = 109 psig
  - Analysis Results
    - The maximum Liner plate strain does not reach the allowable limit strain values based on ASME CC 3720.

NON-PROPRIETARY

3.8 Design of Category I Structures

3.8.1 Concrete Containment

- **Ultimate Pressure Capacity (UPC)**
  - Ultimate pressure capacity assessment
    - Strain limit for pressure capacity in accordance with RG 1.216
    - Design-basis accident temperature = 290 ° F
    - **UPC of containment = 158 psig**



Strain contour of tendon, rebar, and liner plate

3.8 Design of Category I Structures

3.8.1 Concrete Containment

- **Structural Design**
  - The results on the design of the flexural reinforcement are summarized in DCD Tables 3.8A-3 and 3.8A-4.
  - For the flexural reinforcement, it is confirmed that the maximum stresses of the provided reinforcement do not exceed the allowable stresses for both the service and factored load conditions.

3.8 Design of Category I Structures

3.8.2 Steel Containment

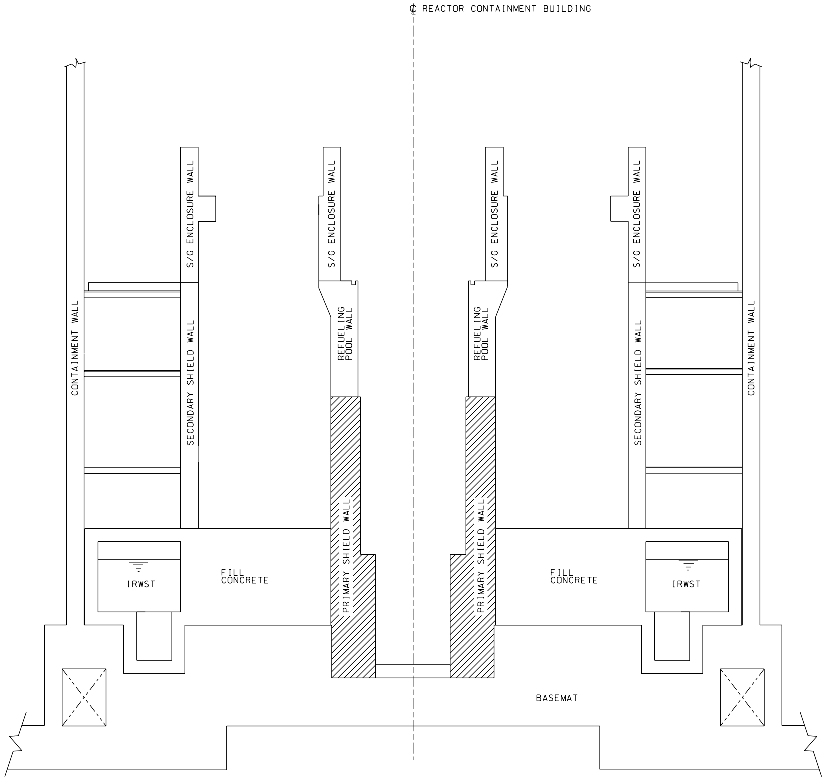
- **Design of the Steel Part of the Containment**
  - The COL applicant is to provide the detailed design results and evaluation of the ultimate pressure capacity of penetrations, including the Equipment Hatch, Personnel Airlocks, Electrical and Piping Penetration.

3.8 Design of Category I Structures

3.8.3 Concrete and Steel Internal Structures of Steel or Concrete Containment

- **Structural Descriptions**

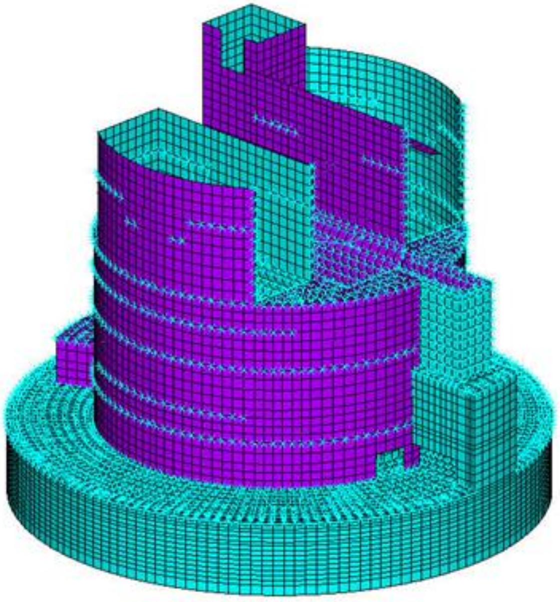
  - The internal structure is a group of reinforced concrete structures that enclose the reactor vessel and primary system
  - The internal structures located in the reactor containment building consist of the following major components:
    - Primary shield wall (PSW)
    - Secondary shield wall (SSW)
    - In-containment refueling water storage tank (IRWST)
    - Operating and intermediate floors
    - Refueling Pool



3.8 Design of Category I Structures

3.8.3 Concrete and Steel Internal Structures of Steel or Concrete Containment

- **Structural Analysis**
  - The 3-dimensional Finite Element models (ANSYS) are developed to perform the structural analysis of the internal structure.



3.8 Design of Category I Structures

3.8.3 Concrete and Steel Internal Structures of Steel or Concrete Containment

- **Structural Design**
  - The results on the design are summarized in DCD Tables 3.8A-21 through 3.8A-25.
  - The concrete section strengths determined from the criteria in ACI 349 are sufficient to resist the design basis loads.

3.8 Design of Category I Structures

3.8.4 Other Seismic Category I Structures

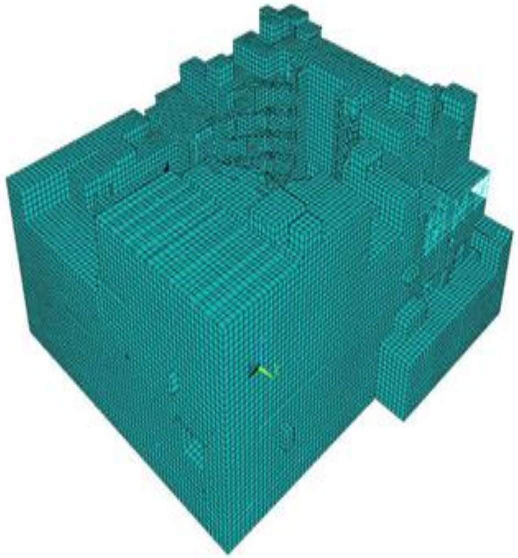
- **Structural Descriptions of AB Superstructure**

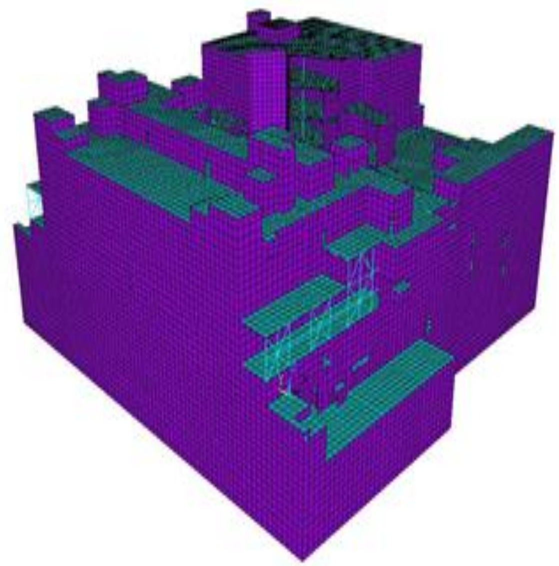
  - AB is reinforced concrete structures, which is composed of rectangular walls, floor slabs, columns, and beams.
  - AB surrounds RCB with a seismic gap of 150 mm (6 in.) and shares a common basemat with RCB. AB structure provides a protection against both external and internal hazards.
  - AB is separated from other buildings by the isolation gap of 900 mm (3 ft.)
  - The auxiliary building is rectangular with maximum dimensions of 106.0 m × 107.6 m (348 ft × 353 ft).

3.8 Design of Category I Structures

3.8.4 Other Seismic Category I Structures

- **Structural Analysis for AB Superstructures**
  - Global Structural Analysis
    - The analysis is to compute all member forces of shear walls and sectional shear forces of slabs in AB.





3.8 Design of Category I Structures

3.8.4 Other Seismic Category I Structures

- **Structural Design of AB Superstructures**
  - The results on the design for **AB** are summarized in DCD Tables 3.8A-29 and 3.8A-33.
  - The concrete section strengths determined from the criteria in ACI 349 are sufficient to resist the design basis loads.

3.8 Design of Category I Structures

3.8.4 Other Seismic Category I Structures

- **Structural Descriptions of EDG Building Block**
  - Emergency diesel generator (EDG) building block consists of two independent buildings:
    - EDG building (EDGB)
    - Diesel Fuel Oil Tank (DFOT) building
  - EDGB is separated from other buildings by the isolation gap of 900 mm (3 ft.)
  - EDGB houses two additional generators, and DFOT building houses the DFOTs thereof.

3.8 Design of Category I Structures

3.8.4 Other Seismic Category I Structures

- **Structural Design of EDG Building Block**
  - The results on the design for EDG Building are summarized in DCD Tables 3.8A-36.
  - The concrete section strengths determined from the criteria in ACI 349 are sufficient to resist the design basis loads.

NON-PROPRIETARY

3.8 Design of Category I Structures

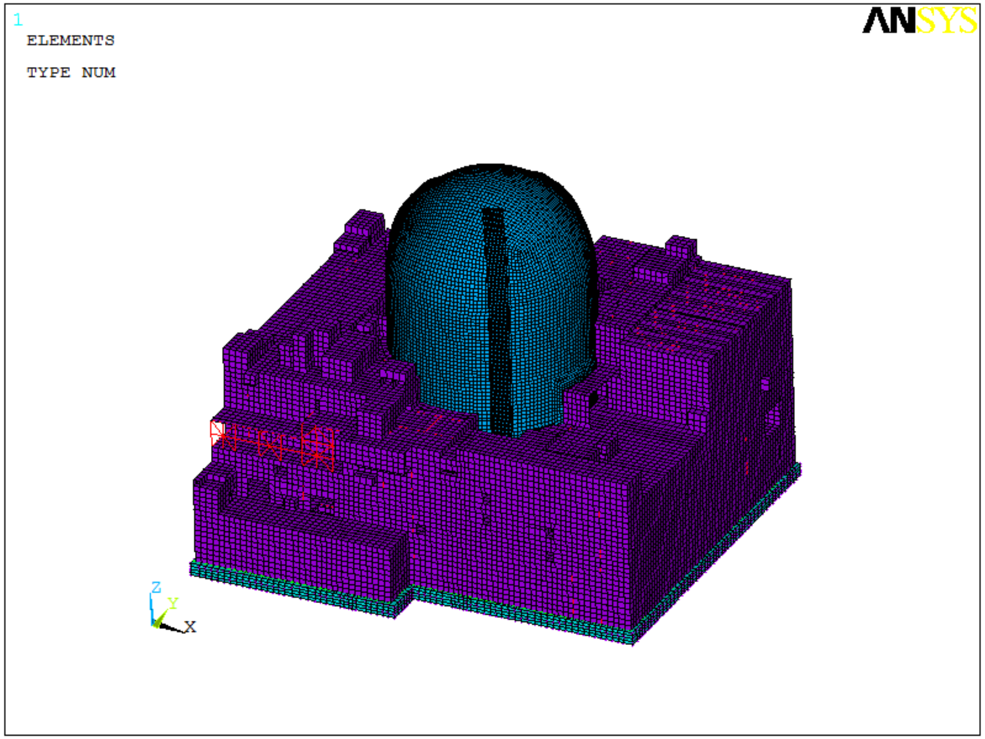
3.8.5 Foundations

- **Structural Descriptions of NI Common Basemat**
  - NI common basemat consists of two areas, one central circular shaped area which supports RCB, and the other rectangular shaped area which supports AB.
  - Disk-shaped reinforced concrete structure in RCB area has variable thickness.

3.8 Design of Category I Structures

3.8.5 Foundations

- **Structural Analysis**
  - For the applied loads on the NI basemat analysis, the equivalent static acceleration method is used to consider the seismic load.



3.8 Design of Category I Structures

3.8.5 Foundations

- **Stability Check Result**
  - Stability check for overturning, sliding, and floatation of NI common basemat was performed in accordance with Section II of SRP 3.8.5 and the results are as follows.

NI Common Basemat		Allowable Criteria (A)	Factor of Safety (B)	Result (A < B)
Overturning	by Wind	1.5	16.46	OK
	by Earthquake	1.1	1.24	OK
Sliding	by Wind	1.5	8.30	OK
	by Earthquake	1.1	1.25	OK
Floatation		1.1	3.39	OK

3.8 Design of Category I Structures

3.8.5 Foundations

- **Structural Design of NI common basemat**
  - The results on the design for RCB Basemat and AB Basemat are summarized in DCD Tables 3.8A-10 through Tables 3.8A-13.
  - The concrete section strengths determined from the criteria in ASME Section III Division 2 Subsection CC for RCB Basemat and ACI 349 for AB Basemat are sufficient to resist the design basis loads.

3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components (1/2)

- This subsection provides the transients used in the design and fatigue analysis of ASME Code Class 1 components and reactor internals.
- The design transients give fluid system pressure, temperature, flow rate, and frequency.
  - Not cover the seismic loading and other mechanical loading on each component.
- The design transient items of each Service Level (A, B, C and D) and test conditions are addressed.
- The 60-year design life is considered when determining the number of occurrences of each transient.
- Fatigue evaluation includes the effects of reactor coolant environment of the APR1400 components
- The frequencies of events traditionally categorized as a Service Level C condition are conservatively modified to be classified as a Service Level B condition for design purpose.

3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components (2/2)

- A number of computer programs, which are commercial codes and in-house codes, are used in the stress and structural analyses for ASME Code Class systems, components and supports.
- All computer programs are verified and validated in accordance with design control methods, consistent with the quality assurance program described in Chapter 17.
- Experimental stress analysis is not used for the APR1400.
- In order to consider the evaluation of the faulted conditions, the major components of the Reactor Coolant System (RCS) are designed to withstand the loads associated with the pipe breaks described in Section 3.6.
- The system or subsystem analysis used to establish or confirm loads that are specified for the design of components and supports is performed on an elastic basis.

3.9 Mechanical Systems and Components

3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment (1/3)

- **Reactor Coolant System (RCS) Structural Analysis (App. 3.9B)**
  - Finite element model is used for RCS Analysis
  - Static Analysis for normal operating conditions
    - Dead weight, Pressure and Temperature
  - Seismic Analysis
    - Described in Section 3.7
  - Postulated pipe break analysis
    - Breaks effects of pipe lines to which LBB concept is not applied
    - Jet impingements and thrust
    - Subcompartment pressure and blowdown loads
    - Branch nozzle loads

3.9 Mechanical Systems and Components

3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment (2/3)

- **Flow-Induced Vibration Assessment for Reactor Internals (3.9.2.3, 3.9.2.4, 3.9.2.6 and APR1400-Z-M-NR-14009)**
  - APR1400 is classified as non-prototype Category I with Palo Verde Unit 1 as the valid prototype in accordance with RG 1.20.
    - APR1400 reactor internals are substantially the same arrangement, design, size and operating conditions as the valid prototype (Palo Verde 1).
  - Analysis Program for Comprehensive Vibration Assessment Program
    - Calculation of hydraulic loads
    - Calculation of dynamic response of reactor internals
    - Measured responses are smaller than the predicted values
  - Inspection Program for Comprehensive Vibration Assessment Program
    - Pre and post hot functional test inspection

3.9 Mechanical Systems and Components

3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment (3/3)

- **Dynamic System Analysis for the Reactor Internals under Faulted Conditions**
  - Dynamic analyses for the reactor internals and core are performed to determine the maximum structural responses under the pipe breaks and seismic loadings.
    - Excitation : Pipe break blowdown loads and reactor vessel motions
    - Structural analysis using finite element method
    - The analysis results for the core are provided to evaluate the structural integrity of fuel assembly as shown in DCD Subsection 4.2.
  - The analysis results for the faulted conditions shows that the reactor internals meet the stress limits of ASME Section III, Subsection NG.
- **Dynamic System Analysis for the CEDM**
  - Calculation of the maximum structural responses to confirm structural integrity of the pressure housings and scramability.

3.9 Mechanical Systems and Components

3.9.3 ASME Code Class 1, 2 and 3 Components, Component Supports, and Class CS Core Support Structures (1/2)

- **Loading Combinations, Design Transients, and Stress Limits**
  - Loading combinations: Categorized as Design, Level A, Level B, Level C, and Level D conditions
  - Design Transients: Design pressure, temperature, and other loading
  - Stress limits: Stress analysis and fatigue evaluations
- **Design and Installation of Pressure Relief Devices**
  - Designed in accordance with ASME Section III, Appendix O.
  - The pressurizer pilot-operated safety relief valves (POSRV) are designed to provide overpressure protection for the RCS.
  - Pressure-relieving devices
    - Class 2 systems: on the steam line and the containment isolation portion of the normal shutdown cooling system (SCS)
    - Class 3 systems: on heat exchangers, tanks, and piping lines to prevent overpressurization of the components and systems.

3.9 Mechanical Systems and Components

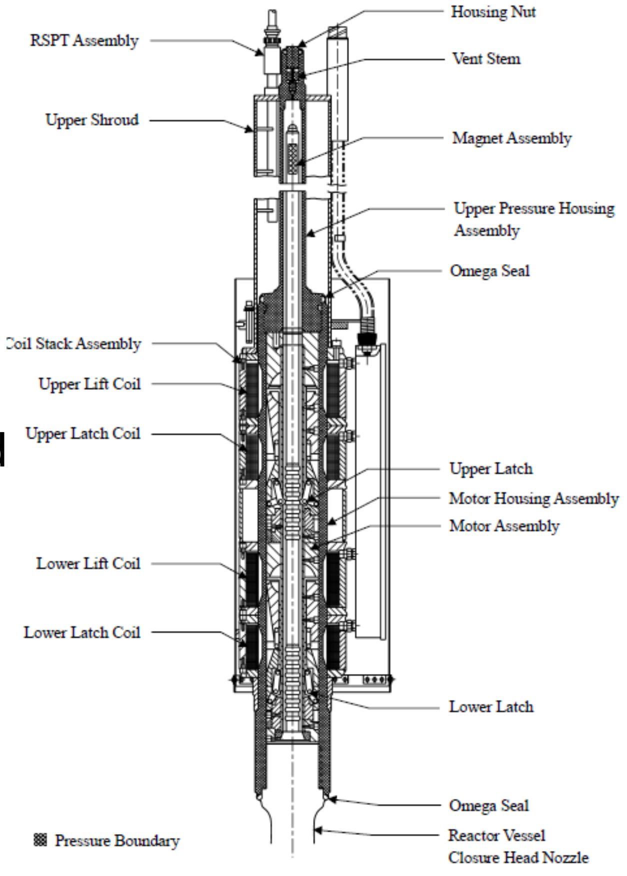
3.9.3 ASME Code Class 1, 2 and 3 Components, Component Supports, and Class CS Core Support Structures (2/2)

- **Pump and Valve Functionality Assurance**
  - The functional design and qualification of safety-related active components are performed in accordance with ASME QME-1-2007.
  - The functional capability of active components during and after exposure to design basis events is confirmed by design, analysis, inspection, testing and startup/periodic in-service testing.
- **Component Support Design**
  - Designed and constructed in accordance with ASME Section III and ASME Code Case(s) approved in NRC RG 1.84.
  - Snubbers, used as shock arrestors for safety-related systems and components, are minimized to the extent practical through the use of design optimization.
  - Reasonable assurance of snubber operability is provided by incorporating analytical, design, installation, in-service, and verification criteria.

3.9 Mechanical Systems and Components

3.9.4 Control Element Drive Mechanisms

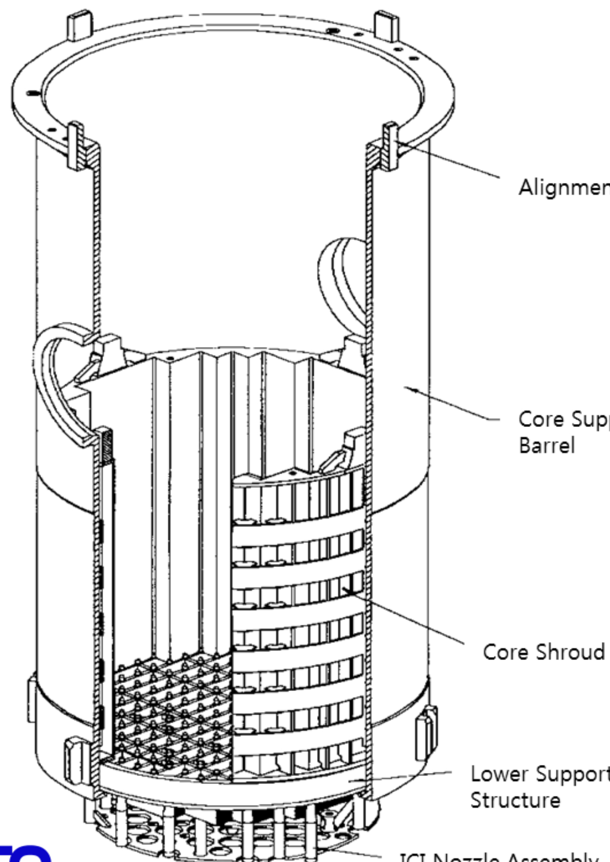
- This subsection provides information on design, functional requirements, and operability assurance program for the Control Element Drive Mechanism (CEDM) portion of the Control Rod Drive System (CRDS).
- The CEDM for the APR1400 is based on the System 80 CEDM which has been used in many nuclear power plants in the United States and Korea.
- The CEDM pressure housing is designed in accordance with the ASME Code Section III, Subsection NB.

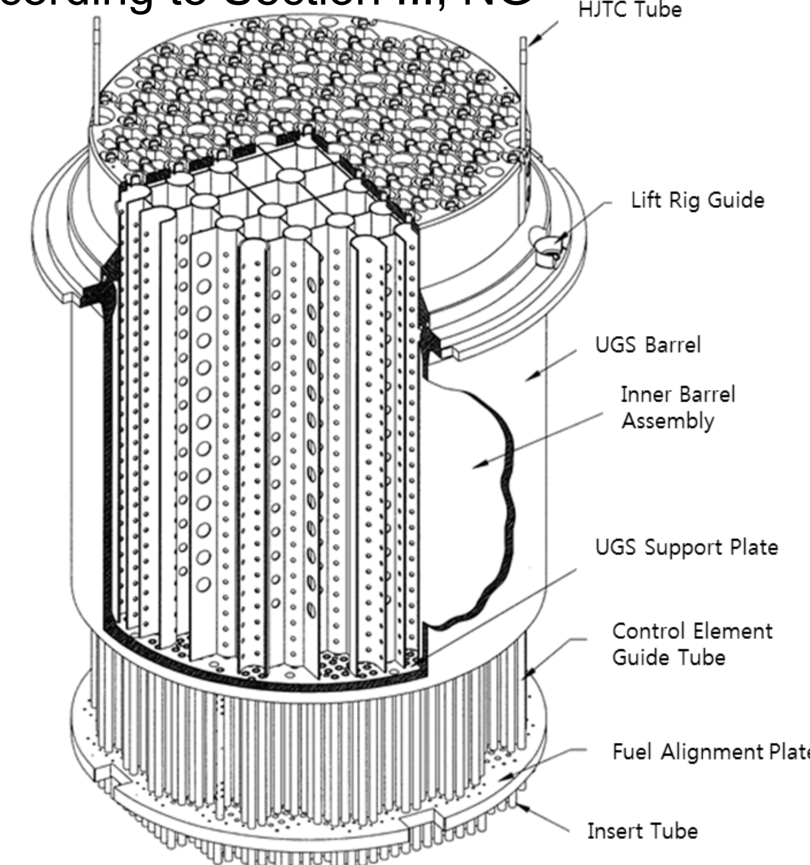


3.9 Mechanical Systems and Components

3.9.5 Reactor Pressure Vessel Internals (1/2)

- Safety Class : Safety Class 3 (ANSI/ANS 51.1)
- Seismic Category : Seismic Category I (RG 1.29)
- Designed and constructed according to Section III, NG

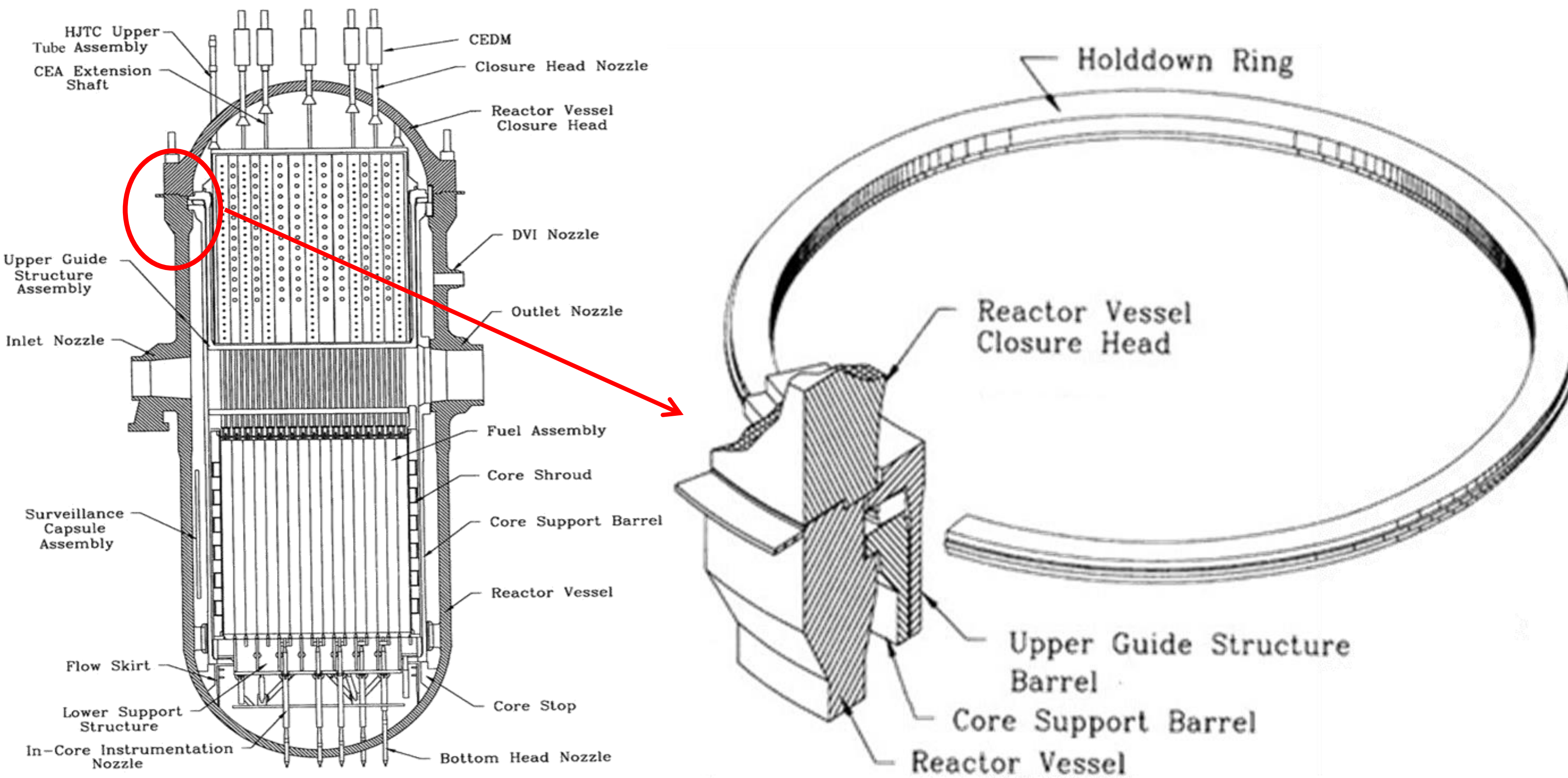




3.9 Mechanical Systems and Components

3.9.5 Reactor Pressure Vessel Internals (2/2)

- To provide axial force on the flange of the upper guide structure barrel and core support barrel.
- To be designed to accommodate the differential thermal expansion.



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3.9 Mechanical Systems and Components

3.9.6 Functional Design, Qualification, and Inservice Testing Program for Pumps, Valves, and Dynamic Restraints

- **Inservice Testing (IST) Program**
  - IST program for ASME Code, Section III, Class 1, 2 and 3 safety-related pumps, valves, and dynamic restraints is developed in accordance with the requirements of ASME OM Code, as required by 10 CFR 50.55a(f) and the acceptable ASME Code Cases listed in RG 1.192 that are incorporated by 10 CFR 50.55a(b).
  - The COL applicant will provide a full description of the IST program.
- **Functional Qualification**
  - Functional qualification of safety-related pumps, valves, and dynamic restraints (snubbers) are performed in accordance with ASME QME-1-2007, as endorsed in RG 1.100, Rev.3.

3.10 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

NON-PROPRIETARY

3.10.1 Seismic Qualification Criteria

- Seismic Qualification Requirements for Seismic Category I instrumentation, electrical equipment, and mechanical equipment::
  - To provide reasonable assurance of structural integrity and performance of their designated safety-related function(s) or intended function(s) under the postulated SSE in combination with other concurrent loading conditions identified in the equipment's design specification.
- IEEE Std 344-2004, as modified by RG 1.100, for safety-related mechanical and electrical equipment and their supports
- IEEE Std. 323-2003 in conjunction with the environmental qualification
- ASME Code, Section III for structural integrity of safety-related pressure boundary components
- ASME QME-1-2007, as modified by RG 1.100, for qualification of active mechanical equipment

NON-PROPRIETARY

3.10.2 Methods and Procedures for Qualifying Mechanical and Electrical Equipment and Instrumentation

- **Testing**
  - Conducted for equipment that cannot be qualified with analysis alone or equipment having components that potentially cause any malfunctions related to their intended functions.
- **Analysis**
  - Analysis without testing is acceptable only if structural integrity alone can assure the design-intended design function.
- **Combination of Testing and Analysis**
  - Utilized when the equipment cannot be practically qualified by analysis or testing alone

NON-PROPRIETARY

3.10.3 Methods and Procedures of Analysis or Testing of Supports of Mechanical and Electrical Equipment and Instrumentation

- **Tests or analyses to assure structural capability**
  - Electrical equipment and instrumentation supports (including instrument racks, control consoles, cabinets, and panels) are tested with the equipment installed or equivalent dummy.
  - For mechanical equipment supports (including pumps, valves, valve operators and fans), in accordance with ASME Code, Section III
  - For instrumentation line supports, using the criteria from ASME Code, Section III, Subsection NF for Equipment Class 1 and 2 supports

3.10 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

3.10.4 Test and Analyses Results and Experience Database

- **Seismic qualification file includes:**
  - Qualification method used for equipment
  - Tests and analyses results
  - List of systems, equipment
  - Equipment support structures
  - Seismic Qualification Summary Data Sheets (SQSDSs), which summarize the component's qualification
  - Seismic Input Requirements

3.11 Environmental Qualification of Mechanical and Electrical Equipment

NON-PROPRIETARY

3.11.1 Equipment Location and Environmental Conditions

- **Equipment Location**
  - Location of each equipment is specified in DCD Table 3.11-2.
  - Classified Mild or Harsh based on the environmental conditions.
- **Environmental Conditions**
  - Normal, AOOs, DBA, Post DBA
  - Environmental parameter values for each room are specified in APR1400 Equipment Qualification Program (APR1400-E-X-NR-14001-P) Table 3.

3.11 Environmental Qualification of Mechanical and Electrical Equipment

NON-PROPRIETARY

3.11.2 Qualification Tests and Analyses

- **Environmental Qualification of Class 1E equipment**
  - Conforms to the requirements of 10 CFR 50.49, 10 CFR 50 Appendix B, NRC RG 1.89, and IEEE Std. 323.
  - Valve actuator(IEEE Std. 382), Cable(IEEE Std. 383), etc.

3.11.3 Qualification Test Results

- **Results and status of qualification are documented in auditable files in accordance with 10 CFR 50.49(j)**

3.11.4 Loss of Ventilation

- **The vital instrument and equipment are served with 100% redundancy of the HVAC unit**
- **Related section: 6.4, 9.4**

3.11 Environmental Qualification of Mechanical and Electrical Equipment

3.11.5 Radiation Environment

- **Radiation Qualification are developed based on:**
  - Up to the time the equipment is required to remain functional
  - Limiting DBAs: LOCA, MSLB, and FHA
- **Assumptions for determining normal/accident condition TIDs**
  - Used codes: Microshield code / RUNT-G code
  - Based on NRC RGs 1.89 and 1.183
  - 1.0% failed fuel for normal TIDs and accident source term used for radiological consequence analysis for accident TIDs
  - 60 years of continuous normal operation with full power plus 1 year post-accident
  - Additional margin of 10% for uncertainty of test

3.11 Environmental Qualification of Mechanical and Electrical Equipment

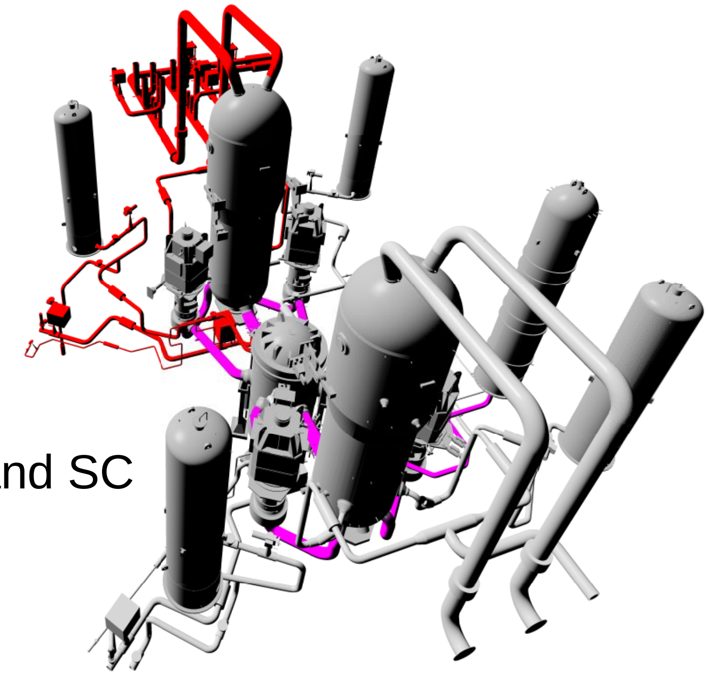
NON-PROPRIETARY

3.11.6 Qualification of Mechanical Equipment

- **Active mechanical equipment**
  - Focus on nonmetallic parts
  - Conforms to ASME QME-1, Appendix QR-B
- **Passive mechanical equipment**
  - Safety function: maintain structural integrity

3.12 Piping Design Review

3.12.1 Introduction



- **This section provides**
  - Structural integrity
  - Functional capability
- **Graded Approach**
  - RCS main loops, Surge line, DVI, and SC
  - MS and FW from nozzle to MSVH

3.12.2 Codes and Standards

- **The safety-related piping system design and analysis**
  - 2007 Edition with 2008 addenda of the ASME Section III
- **Pipe supports**
  - NF of the 2007 Edition with 2008 addenda of the ASME Section III

3.12 Piping Design Review

3.12.3 Piping Analysis Methods

- Procedure for analytical modeling, selection of frequencies, damping criteria, combination of modal responses
- Seismic analysis methods
  - Response spectrum, Time-history, or Equivalent static load method

3.12.4 Piping Modeling Technique

- Computer Codes : PIPESTRESS, ANSYS, RELAP5, GTSTRUDL
- The piping benchmark problems prescribed in NUREG/CR-1677 are used to validate the PIPESTRESS computer program used in piping system analysis.

3.12.5 Piping Stress Analysis Criteria

- Based on the methodology and equations from the ASME Code, pipe stresses are calculated for various load combinations. The ASME Code includes design limits for design conditions; Service Levels A, B, C, D; and testing.
- The environmental fatigue for class 1 piping is performed in accordance with NRC RG 1.207

3.12 Piping Design Review

3.12.6 Piping Support Design Criteria

- Seismic Category I pipe supports are designed in accordance with ASME Section III, NF for Service Levels A, B, C, and D.
- For non-seismic category pipe supports supporting piping analyzed to ASME B31.1, the requirements of ASME B31.1 are met, where applicable
- Baseplate and Anchor Bolt Design for Piping Support
  - ACI 349-01 Appendix B, NRC RG 1.199, NRC Bulletin 79-02

3.13 Threaded Fasteners (ASME Section III Class 1, 2, and 3)

3.13.1 Design Consideration (1/2)

- **Materials Selection**
  - Selected in accordance with ASME Section III NCA-1220 and NB/NC/ND-2128
  - Fabricated using the materials prescribed in ASME Section III or Code cases allowed by RG 1.84
  - Prevention of boric acid corrosion (BAC), galvanic corrosion unless considered to be acceptable
- **Special Materials Fabrication Processes and Special Controls**
  - in accordance with Section II, Section III NB/NC/ND-2200
  - Cleaned in accordance with RG 1.28
- **Fabrication Inspection**
  - Inspected in accordance with ASME Section III, NB/NC/ND-2580

3.13 Threaded Fasteners (ASME Section III Class 1, 2, and 3)

3.13.1 Design Consideration (2/2)

- **Lubricants**
  - Selected in accordance with NUREG-1339
  - Acceptable lubricants : Loctite N-5000, Neolube, and Never Seez Pure Nickel Special Nuclear Grade
  - MoS_2 is not allowed
- **Fracture Toughness for Ferritic Threaded Fasteners:**
  - Tested in accordance with Section III NB/NC/ND-2300 and 10CFR50 Appendix G
- **Certified Material Test Reports**
  - Quality records are controlled, maintained, and stored in accordance with 10CFR Part 50, Appendix B; ASME NQA-1; and ASME NCA-3860

NON-PROPRIETARY

3.13 Threaded Fasteners (ASME Section III Class 1, 2, and 3)

3.13.2 Inservice Inspection Requirements

- **Preservice Inspection (PSI) and Inservice Inspection (ISI)**
  - The relevant requirements of ASME Section XI are followed
  - Inspection programs are to be submitted to the NRC by the COL applicant

Attachments

❖ RAI Summary

	No. of Questions	No. of Responses	Not Responded	No. of Open Items
3.1	0	0	0	0
3.2	12	12	0	0
3.3	0	0	0	0
3.4	10	10	0	1
3.5	21	21	0	0
3.6	19	14	5	5
3.7	30	30	0	1
3.8	56	56	0	7
3.9	58	58	0	5
3.10	9	9	0	1
3.11	24	24	0	1
3.12	18	18	0	2
3.13	0	0	0	0
Total	257	252	5	23

Chapter 3 Open Items Summary

❖ RAI 8197, Question 03.07.02-1

- Date of issue: Aug. 31, 2015
- Description of issue: Adequacy of the number of modes used in HRHF incoherency analysis to capture the incoherent-motion
- Point of argument:
  - Justification for implementing ISRS reduction levels in excess of those provided in SRP 3.7.2
  - Justification for selection of appropriate number of modes to be used in HRHF incoherency analysis
  - Structural evaluation for the HRHF input motions
  - Justification of non-converged 16 mode response and evaluation of seismic demand on the plant structures for 16 mode response

Chapter 3 Open Items Summary

❖ RAI 8245, Question 03.08.03-5

- Date of issue: Sept. 14, 2015
- Description of issue: Design of concrete and steel internal structures of concrete containment
- Point of argument: Design of concrete slab, Effect on design of internal concrete structure due to separated slab analysis model.

Chapter 3 Open Items Summary

❖ RAI 8285, Question 03.08.05-7

- Date of issue: Oct. 19, 2015
- Description of issue: Construction sequence analysis & settlement of basemat
- Point of argument: Effect on the design of super structure due to construction sequence analysis and allowable settlement criteria

❖ RAI 8285, Question 03.08.05-8

- Date of issue: Oct. 19, 2015
- Description of issue: Liner & non-liner analysis considering 100-40-40 method
- Point of argument: Design of NI common basemat under linear & non-linear analysis considering 100-40-40 method

Chapter 3 Open Items Summary

❖ RAI 8285, Question 03.08.05-12

- Date of issue: Oct. 19, 2015
- Description of issue: Soil bearing pressure
- Point of argument: Comparison of soil bearing pressure between SASSI and structural analysis considering equivalent static acceleration.

❖ RAI 8285, Question 03.08.05-13

- Date of issue: Oct. 19, 2015
- Description of issue: Consideration of crane load for seismic category I structures.
- Point of argument: Technical basis for consideration of crane load

Chapter 3 Open Items Summary

❖ RAI 8285, Question 03.08.05-16

- Date of issue: Oct. 19, 2015
- Description of issue: Soil bearing pressure evaluation
- Point of argument: Explanation of soil bearing pressure evaluation in related documents.

❖ RAI 8285, Question 03.08.05-17

- Date of issue: Oct. 19, 2015
- Description of issue: Differential settlement
- Point of argument: The differential settlement criteria and approaching method to calculate differential settlement, all issues are closed and defer to RAI 8285, Question 03.08.05-7.

Chapter 3 Open Items Summary

❖ RAI 7955, Question 03.12-2

- Date of issue: Jun. 17, 2015
- Description of issue: Evaluations of piping systems selected for the graded approach
- Point of argument: - The evaluations will be finished at the end of June, since the new version of PIPESTRESS was released at the end of 2016.

❖ RAI 8278, Question 03.12-9

- Date of issue: Nov. 16, 2015
- Description of issue: HRHF evaluation of the piping systems within the graded approach
- Point of argument: - Justification for not having evaluated the piping that was selected in the graded approach for HRHF seismic effects
  - The results of evaluations is submitted in May.

Chapter 3 Open Items Summary

❖ RAI 8373, Question 03.12-10

- Date of issue: Nov. 16, 2015
- Description of issue: Time history analysis
- Point of argument: - ASCE standard is not completely consistent with current NRC guidance and staff positions
  - The number of modes is sufficient to ensure that inclusion of all remaining modes does not result in more than a 10 percent increase in the total response of interest.

Attachment: List of COL Items for Ch. 3 (1/14)

COL Identifier	Description
COL 3.3(1)	The COL applicant is to demonstrate that the site-specific design wind speed is bounded by the design wind speed of 64.8 m/s (145 mph).
COL 3.3(2)	The COL applicant is to demonstrate that the site-specific seismic Category II structures adjacent to the seismic Category I structures are designed to meet the provisions described in Subsection 3.3.1.2.
COL 3.3(3)	The COL applicant is to perform an analysis if the site-specific wind and tornado/hurricane characteristics are not bounded by the site parameter postulated for the certified design.
COL 3.3(4)	The COL applicant is to provide reasonable assurance that site-specific structures and components not designed for the extreme wind loads do not impact either the function or integrity of adjacent seismic Category I SSCs.

Attachment: List of COL Items for Ch. 3 (2/14)

COL Identifier	Description
COL 3.4(1)	The COL applicant is to provide the site-specific design of plant grading and drainage.
COL 3.4(2)	The COL applicant is to provide site-specific information on protection measures for the design basis flood, such levees, seawalls, flood walls, revetments or breakwaters or site bulkheads pursuant to RG 1.102 as required in Subsection 2.4.10.
COL 3.4(3)	The COL applicant is to establish procedures and programmatic controls to ensure the availability of the floor drainage.
COL 3.4(4)	The COL applicant is to periodically inspect watertight doors and the penetration seals to ensure their functionality.
COL 3.4(5)	The COL applicant is to provide flooding analysis with flood protection and mitigation features from internal flooding for the CCW Heat Exchanger Building and ESW Building.
COL 3.4(6)	The COL applicant is to provide the site-specific flooding hazards from engineered features, such as water tank collapsing, water piping breaking, etc.
COL 3.4(7)	The COL applicant is to confirm that the potential site-specific external flooding events are bounded by design basis flood values or otherwise demonstrate that the design is acceptable.
COL 3.4(8)	The COL applicant is to provide the site-specific dewatering system if the plant is built below the design basis flood level.
COL 3.4(9)	The COL applicant is to describe the basis for the Probable Maxium Flood (PMF) to determine the maximum site-specific ground water elevation above the grade that may occur from tsunami or hurricane sources.
COL 3.4(10)	The COL applicant is to identify any site-specific physical models that could be used to predict prototype performance of hydraulic structures and systems.

Attachment: List of COL Items for Ch. 3 (3/14)

COL Identifier	Description
COL 3.5(1)	The COL applicant is to provide the procedure for heavy load transfer to strictly limit the transfer route inside and outside containment during plant maintenance and repair periods.
COL 3.5(2)	The COL applicant is to provide the procedures which ensure that equipment required during maintenance, should be removed from containment prior to operation, moved to a location where it is not a potential hazard to SSC important to safety, or seismically restrained.
COL 3.5(3)	The COL applicant is to perform an assessment of the orientation of the turbine generator of this and other unit(s) at multi-unit sites for the probability of missile generation using the evaluation of Subsection 3.5.1.3.2 to verify that essential SSCs are outside the low-trajectory turbine missile strike zone.
COL 3.5(4)	The COL applicant is to evaluate site-specific hazards induced by external events that may produce more energetic missiles than tornado or hurricane missiles, and provide reasonable assurance that seismic Category I and II structures are designed to withstand these loads.
COL 3.5(5)	The COL applicant is to confirm that automobile missiles cannot be generated within a 0.5 mile radius of safety-related SSCs that would lead to impact higher than 10.06 m (33 ft) above plant grade.
COL 3.5(6)	The COL applicant is to identify applicable tornado missile spectra and associate velocities for the compound building, and to evaluate the missile protection provided by the building.
COL 3.5(7)	The COL applicant is to evaluate the potential for site proximity explosions and missiles due to train explosions (including rocket effects), truck explosions, ship or barge explosions, industrial facilities, pipeline explosions, or military facilities.
COL 3.5(8)	The COL applicant is to provide justification for the site-specific aircraft hazard and an aircraft hazard analysis in accordance with the requirements of NRC RG 1.206.
COL 3.5(9)	The COL applicant is to provide reasonable assurance that site-specific structures and components not designed for missile loads will not prevent safety-related SSCs from performing their safety function.

Attachment: List of COL Items for Ch. 3 (4/14)

COL Identifier	Description
COL 3.6(1)	The COL applicant is to identify the site-specific SSCs that are safety related or required for safe shutdown that are located near high- and moderate-energy piping systems and that are susceptible to the consequences of piping failures.
COL 3.6(2)	The COL applicant is to provide a list of site-specific high- and moderate-energy piping systems including layout drawings and protection features and the failure modes and effects analysis for safe shutdown due to the postulated HELBs.
COL 3.6(3)	The COL applicant is to confirm that the bases for the LBB acceptance criteria are satisfied by the final as-built design and materials of the piping systems as site-specific evaluations, and is to provide the information including LBB evaluation report for the verification of LBB analyses.
COL 3.6(4)	The COL applicant is to provide the procedure for initial filling and venting to avoid the known causes for water hammer in each piping system designed for LBB.
COL 3.6(5)	The COL applicant is to provide the information on welding of Alloy 52/52M/152 concerning the residual stress and dilution effects of welds.

Attachment: List of COL Items for Ch. 3 (5/14)

COL Identifier	Description
COL 3.7(1)	The COL applicant is to demonstrate the applicability of soil degradation models used in site-specific site response analysis for the site conditions.
COL 3.7(2)	The COL applicant is to compare the site-specific strain-compatible soil properties with generic soil properties in order to confirm that the site meets the generic soil profile used in the standard design.
COL 3.7(3)	The COL applicant is to provide the seismic design of the seismic Category I SSCs and seismic Category II structures that are not part of the APR1400 standard plant design. The seismic Category I and II structures are as follows: a. Seismic Category I essential service water building b. Seismic Category I component cooling water heat exchanger building c. Seismic Category II turbine generator building d. Seismic Category II compound building e. Seismic Category II alternate alternating current gas turbine generator building
COL 3.7(4)	The COL applicant is to confirm that any site-specific non-seismic Category I structures are designed not to degrade the function of a seismic Category I SSC to an unacceptable safety level due to their structural failure or interaction. The COL applicant is to confirm that the calculated relative displacements do not exceed the gaps between seismic Category I and non-seismic Category I structures.

Attachment: List of COL Items for Ch. 3 (6/14)

COL Identifier	Description
COL 3.7(4)	The COL applicant is to apply the site-specific FIRS as seismic input motions and to establish a site-specific soil profile as a supporting media for the seismic analysis of the seismic Category II structures. The COL applicant is to apply the same seismic analysis procedure as the seismic Category I structures to the seismic Category II structures. The COL applicant is to perform the structural design of the seismic Category II structures using the design codes described in Subsection 3.7.2.8 and Table 3.2-1. The COL applicant is to check the potential effects of sliding and uplift for the seismic Category II structures using the same approach applied in the stability check for the seismic Category I structures.
COL 3.7(5)	The COL applicant is to perform any site-specific seismic design for dams that is required.
COL 3.7(6)	The COL applicant is to perform seismic analysis of buried seismic Category I conduits, and tunnels.
COL 3.7(7)	The COL applicant is to perform seismic analysis for the seismic Category I above-ground tanks.
COL 3.7(8)	The COL applicant that references the APR1400 design certification will determine whether essentially the same seismic response from a given earthquake is expected at each unit in a multi-unit site or each unit is to be provided with a separate set of seismic instruments.
COL 3.7(9)	The COL applicant is to confirm details of the locations of the triaxial time-history accelerographs.
COL 3.7(10)	The COL applicant is to identify the implementation milestones for the seismic instrumentation implementation program based on the discussion in Subsections 3.7.4.1 through 3.7.4.5.
COL 3.7(11)	The COL applicant is to prepare a procedure for the post shutdown inspection and plant restart in accordance with the guidance of NRC RG 1.167.

Attachment: List of COL Items for Ch. 3 (7/14)

COL Identifier	Description
COL 3.8(1)	The COL applicant is to perform concrete long-term material testing in a way which verifies physical properties of materials used during the design stage and the characteristics of long term deformation of concrete.
COL 3.8(2)	The COL applicant is to provide the detailed design results and evaluation of the ultimate pressure capacity of penetrations, including the equipment hatch, personnel airlocks, electrical and piping penetration in accordance with RG 1.216.
COL 3.8(3)	The COL applicant is to provide detailed analysis and design procedure for the equipment hatch, personnel airlocks, and electrical penetrations.
COL 3.8(4)	The COL applicant is to provide a detailed analysis and design procedure for the transfer tube assembly.
COL 3.8(5)	The COL applicant is to provide the design of site-specific seismic Category I structures such as the essential service water building and the component cooling water heat exchanger building, essential service waterconduits, component cooling water piping tunnel, and class 1E electrical duct runs.
COL 3.8(6)	The COL applicant is to evaluate any applicable site-specific loads such asexplosive hazards in proximity to the site, projectiles and missiles generated from activities of nearby military installations, potential nonterrorism related aircraft crashes, and the effects of seiches, surges, waves, and tsunamis.
COL 3.8(7)	The COL applicant is to perform the analysis and design of the steel plate for the new fuel storage pit.
COL 3.8(8)	The COL applicant is to determine the environmental condition associated with the durability of concrete structures and provide the concrete mix design that prevents concrete degradation including the reactions of sulfate and other chemicals, corrosion of reinforcing bars, and influence of reactive aggregates.

Attachment: List of COL Items for Ch. 3 (8/14)

COL Identifier	Description
COL 3.8(9)	The COL applicant is to determine construction techniques to minimize the effects of thermal expansion and contraction due to hydration heat, which could result in cracking.
COL 3.8(10)	For safety and serviceability of seismic Category I structures during the operation of the plant, the COL applicant is to provide appropriate testing and inservice inspection programs to examine the condition of normally inaccessible, below-grade concrete for signs of degradation and to conduct periodic site monitoring of ground water chemistry. Inservice inspection of the accessible portion of concrete structures is also to be performed.
COL 3.8(11)	The COL applicant is to verify that the coefficient of friction between the lean concrete and waterproofing membrane is bounded by 0.55.
COL 3.8(12)	The COL applicant is to provide reasonable assurance that the design criteria listed in Table 2.0-1 are met or exceeded.
COL 3.8(13)	The COL applicant is to verify that the coefficient of friction between the lean concrete and the supporting medium at the site is equal to or higher than 0.55.
COL 3.8(14)	The COL applicant is to confirm that uneven settlement due to construction sequence of the NI basemat falls within the values specified in Table 2.0-1.
COL 3.8(15)	The COL applicant is to provide a site-specific monitoring program and to monitor differential settlement, tilt, and angular distortion are bounded by following values during construction and plant operation. Allowable differential settlement associated with tilt: 1/1200 Allowable differential settlement associated with angular distortion: 1/750
COL 3.8(16)	The COL applicant is to provide testing and inservice inspection program to examine inaccessible areas of the concrete structure for degradation and to monitor groundwater chemistry.

Attachment: List of COL Items for Ch. 3 (9/14)

COL Identifier	Description
COL 3.8(17)	The COL applicant is to provide the following soil information for the APR1400 site: 1) elastic shear modulus and Poisson's ratio of the subsurface soil layers, 2) consolidation properties including data from one-dimensional consolidation tests (initial void ratio, Cc, Ccr, OCR, and complete e-log p curves) and time-versus-consolidation plots, 3) moisture content, Atterberg limits, grain size analyses, and soil classification, 4) construction sequence and loading history, and 5) excavation and dewatering programs.

Attachment: List of COL Items for Ch. 3 (10/14)

COL Identifier	Description
COL 3.9(1)	The COL applicant is to provide the inspection results for the APR1400 reactor internals classified as non-prototype Category I in accordance with NRC RG 1.20.
COL 3.9(2)	The COL applicant is to provide a summary of the maximum total stress, deformation, and cumulative usage factor values for each of the component operating conditions for ASME Code Class 1 components except for ASME Code Class 1 nine major components. For those values that differ from the allowable limits by less than 10 percent, the contribution of each loading category (e.g., seismic, deadweight, pressure, and thermal) to the total stress is provided for each maximum stress value identified in this range. The COL applicant is to also provide a summary of the maximum total stress and deformation values for each of the component operating conditions for Class 2 and 3 components required to shut down the reactor or mitigate consequences of a postulated piping failure without offsite power (with identification of those values that differ from the allowable limits by less than 10 percent).
COL 3.9(3)	The COL applicant is to identify the site-specific active pumps.
COL 3.9(4)	The COL applicant is to provide an IST program including the type of testing and frequency of site-specific pumps subject to IST in accordance with the ASME Code.
COL 3.9(5)	The COL applicant is to provide an IST program including the type of testing and frequency of site-specific valves subject to IST in accordance with the ASME Code.
COL 3.9(6)	The COL applicant is to provide a table listing all safety-related components that use snubbers in their support systems.

Attachment: List of COL Items for Ch. 3 (11/14)

COL Identifier	Description
COL 3.10(1)	The COL applicant is to provide documentation that the designs of seismic Category I SSCs are analyzed for OBE, if OBE is higher than 1/3 SSE.
COL 3.10(2)	The COL applicant is to investigate if site-specific spectra generated for the COLA exceed the APR1400 design spectra in the high-frequency range. Accordingly, the COL applicant is to provide reasonable assurance if the functional performance of vibration-sensitive components in the high frequency range.
COL 3.10(3)	The COL applicant is to develop the equipment seismic qualification files that summarize the component's qualification, including a list of equipment classified as seismic Category I in Table 3.2-1 and SQSDS for each piece of seismic Category I equipment.
COL 3.10(4)	The COL applicant is to perform equipment seismic qualification for seismic Category I equipment an provide milestones and completion dates of equipment seismic qualification program.

Attachment: List of COL Items for Ch. 3 (12/14)

COL Identifier	Description
COL 3.11(1)	The COL applicant is to identify and qualify the site-specific mechanical, electrical, I&C, and accident monitoring equipment specified in RG 1.97.
COL 3.11(2)	The COL applicant is to identify the nonmetallic parts of mechanical equipment in procurement process.
COL 3.11(3)	The COL applicant is to address operational aspects for maintaining the environmental qualification status of components after initial qualification.
COL 3.11(4)	The COL applicant is to provide a full description of the environmental qualification of mechanical and electrical equipment program.
COL 3.11(5)	The COL applicant is to document the qualification test results and qualification status in an auditable file for each type of equipment in accordance with the requirements 10 CFR 50.49(j).
COL 3.11(6)	The COL applicant is to describe the EQP implementation milestones based on the APR1400 EQP.
COL 3.11(7)	The COL applicant is to provide room number designation for those unidentified rooms in Table 3.11-2

Attachment: List of COL Items for Ch. 3 (13/14)

COL Identifier	Description
COL 3.12(1)	If COL applicant finds it necessary to route ASME Class 1, 2, or 3 piping systems outside the structure, the wind and/or tornado load must be included in the plant design basis loads considering the site-specific loads.

Attachment: List of COL Items for Ch. 3 (14/14)

COL Identifier	Description
COL 3.13(1)	The COL applicant is to maintain quality assurance records including CMTRs on ASME Section III Class 1, 2, and 3 component threaded fasteners in accordance with the requirements of 10 CFR 50.71.
COL 3.13(2)	The COL applicant is to submit the preservice and inservice inspection programs for ASME Section III Class 1, 2, and 3 component threaded fasteners to the NRC prior to performing the inspections.

Attachment: Acronyms (1/4)

- AB: Auxiliary Building
- ACI: American Concrete Institute
- ADAMS:Agency Wide Documents Access and Management System
- AISC: American Institute of Steel Construction
- ANS: American Nuclear Society
- ANSI: American National Standards Institute
- APR1400:Advanced Power Reactor 1400
- ASCE:American Society of Civil Engineers
- AOO: Anticipated Operational Occurrence
- ASCE: American Society of Civil Engineers
- ASME: American Society of Mechanical Engineers
- BAC:Boric Acid Corrosion
- BTP:Brach Techniacal Position
- CCW:Component Cooling Water
- CEDM:Control Element Drive Mechanism
- CFR: Code of Federal Regulations
- CSDRS:Certified Seismic Design Response
- COL: Combined License
- COLA: Combined License Applicant

Attachment: Acronyms (2/4)

- CRDS: Control Rod Drive System
- DBA: Design Basis Accident
- DCD: Design Control Document
- DFOT: Diesel Fuel Oil Tank
- DVI: Direct Vessel Injection
- EDG: Emergency Diesel Generator
- EOL: Emergency Overflow Line
- EPRI: Electric Power Research Institute
- EQ: Equipment Qualification
- ESW: Essential Service Water
- FHA: Fuel Handling Area
- FIRS: Foundation Input Response Spectra
- FW: Feed Water
- GDC: General Design Criteria
- GMRS: Ground Motion Response Spectra
- HELB: High-Energy Line Break
- HRHF: Hard Rock High Frequency
- HVAC: Heating, Ventilation, Air Conditioning
- HVT: Holdup Volume Tank

Attachment: Acronyms (3/4)

- IEEE: Institute of Electrical and Electronics Engineers
- ISI: Inservice Inspection
- IST: Inservice Testing
- IRWST: In-containment Refueling Water Storage Tank
- ISRS: In-Structure Response Spectra
- LBB: Leak Before Break
- LOCA: Loss of Coolant Accident
- MS: Main Steam
- MSLB: Main Steam Line Break
- MSVH: Main Steam Valve House
- NI: Nuclear Island
- NRC: U.S. Nuclear Regulatory Commission
- OBE: Operating Basis Earthquake
- PED: Piping Evaluation Diagram
- PGA: Peak Ground Acceleration
- PMF: Probable Maximum Flood
- POSRV: Pilot Operated Safety Relief Valve
- PSI: Preservice Inspection
- PSW: Primary Shield Wall

Attachment: Acronyms (4/4)

- PWR: Pipe Whip Restraints
- RAI: Request for Additional Information
- RCB: Reactor Containment Building
- RCS: Reactor Coolant System
- RG: Regulatory Guide
- RV: Reactor Vessel
- SC: Shutdown Cooling
- SEI: Structural Engineering Institute
- SCS: Shutdown Cooling System
- SQSDS: Seismic Qualification Summary Data Sheets
- SRP: Standard Review Plan
- SRSS: Square Root of the Sum of the Squares
- SSC: System, Structural and Component
- SSE: Safe Shutdown Earthquake
- SSI: Soil-Structure Interaction
- SSSI: Structure-Soil-Structure Interaction
- SSW: Secondary Shield Wall
- TID: Total Integrated Dose
- UPC: Ultimate Pressure Capacity

Presentation to the ACRS Subcommittee

Korea Hydro & Nuclear Power Co., Ltd (KHNP) and Korea Electric Power Corporation (KEPCO)

APR1400 Design Certification Application Review

Safety Evaluation Report with Open Items: Chapter 3

Design of Structures, Systems, Components, and Equipment

June 5, 2017

Staff Review Team

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

- **Technical Staff**

  - **Jinsuo Nie**, Structural Engineer, Structural Engineering Branch
  - **Robert Roche-Rivera,** Structural Engineer, Structural Engineering Branch
  - **Vaughn Thomas,** Structural Engineer, Structural Engineering Branch
  - **Ata Istar**, Structural Engineer, Structural Engineering Branch
  - **David Heeszel,** Geophysicist, Geoscience and Geotechnical Engineering Branch
  - **Yuken Wong,** Senior Mechanical Engineer, Mechanical Engineering Branch
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Staff Review Team

- **Technical Staff**
  - **Edward Stutzcage,** Health Physicist, Radiation Protection and Accident Consequences Branch
- **Project Managers**
  - **Bill Ward,** Lead Project Manager
  - **Tomeka Terry**, Chapter Project Manager

APR1400 Seismic Analysis and Design Review Approach

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- Conducted pre-application quality assurance audit
- Performed regulatory audit at the applicant facility
- Held bi-weekly public meetings (teleconferences or meetings) to facilitate the resolution of technical issues and to refine RAIs
- Integrated lessons learned from previous DC applications
- Compiled “list of clarification Issues” for minor findings (e.g., inconsistency, verification/justification needed, typo)
- Issued RAIs for significant staff findings (e.g., technical issue, incompleteness)
- Confirmed the methods and techniques used are appropriate
- Ensured consistency with other design certifications

Seismic Parameters & Analysis
SER Section 3.7

Robert Roche-Rivera

Seismic Parameters & Analysis

United States Nuclear Regulatory Commission
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- Reviewed DCD Section 3.7, Appendices 3.7A and 3.7B, and six referenced technical reports
- Held bi-weekly public meetings with the applicant to discuss technical issues
- Performed independent confirmatory analysis of the acceleration time histories
- Conducted a seismic design audit to verify and investigate the implementation of the seismic design criteria, key calculations, and the safety significance of methods differing from SRP guidance
- Confirmed consistency with guidance in SRP, RGs, and referenced seismic standards
- With the exception of one open item, concluded that the seismic design parameters and seismic analysis procedures and criteria delineated by the applicant provides an acceptable basis for the seismic design

Seismic Design Parameters SER Section 3.7.1 and Seismic Subsystem Analysis SER Section 3.7.3

Jinsuo Nie

Certified Seismic Design Response Spectra (CSDRS)

- Both the CSDRS and the CSDRS at the foundation level in the free field ($CSDRS_{ff}$) are required to bound the minimum required response spectrum (MRRS) anchored to 0.1 g
- The envelope of $CSDRS_{ff}$ for all soil profiles, not the $CSDRS_{ff}$ for each soil profile, was initially used to compare with MRRS
- $CSDRS_{ff}$ for soil profiles S6 and S7 show large dips, which the applicant explained later were due to a soil layer interface close to the bottom of the foundation
- However, transfer functions from the ground surface to the outcrop foundation level did not appear to explain these large dips
- During the audit, the transfer functions were found to be from the outcrop bedrock to the outcrop foundation level
- RAI response and DCD markup were updated accordingly

Development of Target PSD

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- Target PSD functions should be properly generated for design response spectra other than RG 1.60 spectral shapes
- APR1400 target PSD function in the horizontal direction below 9 Hz is specified as the same as the one for the RG 1.60 horizontal spectrum
- APR1400 target PSD function in the vertical direction was developed based on one-time scaling of the horizontal target PSD function
- Applicant showed that the PSD functions were in general agreement with those generated based on 30 simulated time histories
- Staff confirmatory analysis showed target PSD functions higher than the applicant's target PSD functions in some frequency ranges
- However, the applicant's minimum target PSD functions are very close to or more conservative than the staff's minimum target PSD functions (Applicant used a factor of 0.8, compared to 0.7 specified in SRP 3.7.1 Appendix B)

Acceleration Time Histories

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- Seed records lack high frequency content and the Fourier phase spectra of some seed records are cyclic, constant, or with a gap in some frequency bands
- The design time histories was found to have high frequency content, and their phase spectra do not show the above issues
- Low-pass filters are applied to the design time histories with corner frequencies below 50 Hz, to remove artificial high frequency contents due to baseline correction and clipping
- However, the effect of the low-pass filters were determined to be insignificant
- The method used to estimate PSD functions is not consistent with the SRP guidance
- The updated PSD functions were shown to exceed the minimum target PSD functions

Seismic System Analysis SER Section 3.7.2

Robert Roche-Rivera

Soil Structure Interaction (SSI) - Sensitivity Studies

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- Original seismic system evaluation lacked sensitivity studies
- The applicant evaluated the effects of:
  - Separation of soil from sidewalls – design basis ISRS for the fixed-base case bound the ISRS for analysis cases considering separation of soil from sidewalls
  - Sensitivity to Poisson's Ratio – analysis cases for several values of Poisson's Ratio confirmed that no numerical instabilities exist in the design basis SSI analysis
  - Basemat Uplift (revised for consistency with SRP guidance) – ground contact ratios are greater than 80 percent, which meets the SRP Section 3.7.2.II.4 criterion for acceptability of linear SSI analysis

Structure-Soil-Structure Interaction Analysis (SSSI)

- Original SSSI evaluation assumed surface-mounted structures;
  - ♦ did not consider pressure distributions on exterior below grade walls due to SSSI effects
- Applicant performed SSSI analysis based on the embedded foundation configuration
- Lateral soil pressures computed from the SSSI and SSI analyses are higher than the dynamic soil pressure originally used in the design of exterior below grade walls in the AB and DFOT Room
- Applicant reevaluated the structural design of exterior below grade walls to consider the calculated maximum lateral soil pressures from the SSSI and SSI analyses

Concrete Containment SER Section 3.8.1, Containment Internal Structures SER Section 3.8.3, and Other Seismic Category I Structures SER Section 3.8.4

Vaughn Thomas

Structural Design of Category I Structures

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- Ensured the applicant's design approach and methodology is reasonable and acceptable
  - ♦ Scope
  - ♦ Level of details
  - ♦ Technical adequacy
- Reviewed DCD Section 3.8 and associated Appendix 3.8A, the listed tables and figures, and the structural design report in accordance with applicable sections of SRP 3.8 acceptance criteria
- Held bi-weekly public meetings with the applicant to discuss technical issues and resolutions for resolving RAIs
- Examined and compared the applicant's results to the applicable code allowable
- Confirmed that the applicant's method for demonstrating the design adequacy of the structures are consistent with the agency's regulatory requirements

Hydrogen Generation Pressure Load

- Reviewed APR1400 DCD Section 3.8.1.3 in accordance with SRP 3.8.1 acceptance criteria
  - ♦ Analyses to demonstrate that the containment structural integrity is maintained when subjected to hydrogen generated pressure loads resulting from fuel cladding and water interaction
  - ♦ Description of the design and analysis procedures not provided
  - ♦ Acceptance criteria not provided
- Applicant performed additional calculation and sensitivity analysis of the 3-D finite element model of the containment structure
- The results demonstrated that the liner strains are below the ASME code allowable limits and the rebar and tendons remain in the elastic range
- The applicant's methodology for demonstrating the structural integrity of the containment subjected to hydrogen pressure load meets the NRC regulatory acceptance criteria
- Provided markup copies that describes the design and analysis approach, and the acceptance criteria for the hydrogen pressure load

Ultimate Pressure Capacity

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- Reviewed APR1400 DCD Section 3.8.1.4 in accordance with SRP 3.8.1 acceptance criteria
  - Design and analysis procedures to determine the UPC of the containment at which structural integrity is retained
  - The applicant's approach and acceptance criteria, used to calculate the UPC of the containment, does is not appear to be in conformance with the approach described in Regulatory Position 1 of RG 1.216
- The design and analysis procedures for determining the UPC are performed in accordance with RG 1.216
- Applicant performed additional calculation and sensitivity analysis of the 3-D finite element model of the containment structure
- The results demonstrated that the rebar, tendon, and liner strains at the ultimate pressure level are below the strain limits
- Included a COL item (COL 3.8(11)) which requires the COL applicant to provide the detailed design results and evaluation of the penetrations

Leak Chase Channels

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- Reviewed APR1400 DCD Section 3.8.3.1 and associated Appendix 3.8A in accordance with SRP 3.8.3 acceptance criteria
  - Design and analysis procedure for the IRWST of the containment internal structures, including leak chase channels and stainless steel liner plate
  - The applicant did not provide descriptions or associated design details of the leak chase channels in the IRWST
- The applicant is committed to using a leak chase channel system to monitor potential leakage of water from the IRWST
- The applicant provided mark copies of applicable section of the DCD that includes the description of the leak chase channel collection system that will be used in the design of the IRWST
- Included a COL item (COL 9.3(5)) for monitoring and inspection of the leak chase channel collection system

Dynamic Lateral Earth Pressure

- Reviewed APR1400 DCD Section 3.8.4.4 and associated Appendix 3.8A in accordance with SRP 3.8.4 acceptance criteria
  - ♦ Design and analysis procedure for the below grade walls in seismic Category I structures considering the governing dynamic lateral earth pressures
  - ♦ The applicant did not adequately describe the approach for developing dynamic earth pressure loads that are considered in the analysis and design of embedded walls
- For the AB and DFOT, the dynamic earth pressures determined from the SSI/SSSI FEA analysis results were higher than those calculated in accordance with ASCE 4-98
- The structural analysis of the AB and the DFOT are revised to use the dynamic earth pressure obtained from the SSI/SSSI analysis as the governing dynamic earth pressure
- The increase in the dynamic earth pressure led to revising the member forces of the exterior walls
- The applicant provided markup copies of applicable sections of the DCD that reflect these changes

Foundations
SER Section 3.8.5

Ata Istar

Tendon Gallery

- APR1400 DCD, Section 3.8.5.1, "Description of the Foundations," and Section 3.8.5.4, "Design and Analysis Procedures"
- SRP 3.8.5, Section I.1.A, description of the arrangement of the gallery and means of either isolating it from the remainder of the base slab or relying on it for some function, such as resisting shears
- SRP 3.8.5, Section II.4.I, description of any unique design features that occur in the load path (e.g., any safety-related function that the tendon gallery may have as part of the function in the prestressed containment)
  - The applicant did not describe the tendon gallery, and the analysis and design approaches used for the tendon gallery
- The applicant provided the description of tendon gallery, and it was included in the analysis and design as part of the NI common basemat

Waterproofing Membrane

- APR1400 DCD, Section 3.8.5.1, “Description of the Foundations”
- SRP 3.8.5, Section I.1.A, if waterproofing membranes are used, the review addresses their effect on the shear resistance of the foundation
  - ♦ The applicant did not provide any description whether waterproofing membranes are used
- The applicant responded that waterproofing membranes will be used for exterior below grade horizontal and vertical surfaces of structures in APR1400 design
- The COL applicant (COL Item 3.8(13)) is to verify that the COF between the lean concrete and waterproofing membrane is greater than 0.55 used in design
- (Q 3.8.5-14 also addresses the smallest COF value between the various potential sliding interfaces in the evaluation of the NI common basemat)

Construction Sequence & Differential Settlements

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- 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-P, "Stability Check for NI Common Basemat."
- SRP 3.8.5, Section I.3. "Load and Load Combinations," "..loads that are induces by construction sequence and the differential settlements..."
- The applicant did not include the superstructure of the RCB and AB in the construction sequence evaluation, and clearly not determine the settlement types of (1) max. vertical settlement, (2) tilt settlement, (3) differential settlement between buildings, and (4) angular distortion throughout the NI foundation.
- Currently, these issues are being addressed by the applicant.

Seismic Instrumentation SER Section 3.7.4

David Heeszel

Seismic Instrumentation

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- KHNP has adequately described
  - ♦ Location of seismic instrumentation
  - ♦ Requirements for development of implementation plan for a COL applicant
- KHNP referenced appropriate Regulatory Guides for
  - ♦ Determining exceedance of the OBE
  - ♦ COL applicant development of plan for plant inspection and restart following a seismic event

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Dynamic Testing and Analysis of SSCs SER Section 3.9.2

Yuken Wong

Dynamic Testing and Analysis of SSCs

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- 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

- Also, reviewed the Comprehensive Vibration Assessment Program (CVAP) report for APR1400 steam generator flow induced vibration and reactor design in comparison with System 80 reactor design such that the Palo Verde Unit I design is prototype reactor and APR1400 is classified as non-prototype Category 1 reactor.
- Staff requested basis for using 33 Hz ZPA – Applicant revised to 50 Hz in DCD
- Staff inquired about tank contents in the analysis – applicant responded hydrodynamic forces exerted by the fluid on the tank walls is in the analysis.
- Staff inquired about the DPVIB computer code benchmarking – Applicant provided acceptable V&V information.

Reactor Pressure Vessel Internals SER Section 3.9.5

Yuken Wong

Reactor Pressure Vessel Internals

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- 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.
- Public meetings were held with the applicant early in the review stage to discuss issues. Many of these issues were addressed by the applicant early. Remaining or additional issues were raised via RAIs to which the applicant responded.
- 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

- Open Item: Concern with guide tube structural integrity. KHNP has since provided valid calculation to show the guide tube structural integrity can be maintained at SSE. Concern is resolved. KHNP was also requested to look into operational history from operating plants with System 80+ design. KHNP investigated Young Gwang Unit 2 since July 1995 and found no failure data on any reactor internals or guide tubes. Staff asked KHNP to check Kori and Palo Verde. Status is ongoing.
- Open Item: Static o-ring at seal table for ICI support system. KHNP provided test data to validate o-ring design. Open item is closed.
- Open Item: Service level conditions that result in IRWST discharge. KHNP provided clarification to which events would result in IRWST discharge. Open item is closed.
- Open Item: Seismic category for reactor internals. KHNP provided clarification that all reactor internals (both core support structures and internal structures) are seismic category I. Confirmatory item.

Seismic & Dynamic Qualification of Equipment SER Section 3.10

Yuken Wong

Seismic & Dynamic Qualification of Equipment

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- Reviewed DCD Tier 2, Section 3.10, 3.7B.7.4, and technical reports
- Verified equipment seismic qualification standards and methods
- Verified procedures to evaluate of effects of hard rock high frequency (HRHF) response spectra
  - For new equipment, qualification will envelop certified seismic design response spectra (CSDRS) and HRHF response spectra
  - Equipment undergone prior qualification
    - Applicant will verify test results envelop CSDRS and HRHF response spectra
    - Applicant will perform screening tests using required response spectra enveloping HRHF response spectra
- Conducted audit of procurement (design) specifications to verify seismic qualification methodologies

Seismic & Dynamic Qualification of Equipment

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- Open item: Design Specification Follow-up Audit items for seismic qualification are acceptable. Applicant will update procurement specifications to address staff audit findings.

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ASME Code Class 1, 2, and 3 Piping Systems and Associated Supports Design SER Section 3.12

Yuken Wong

Piping Analysis and Supports

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- Confirmed APR1400 piping and supports analysis is in accordance with NRC guidance.
- 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.
- Identified that the environmentally assisted fatigue for the reactor coolant loop piping had not been performed. It has since been completed and is currently under review.
- Questioned the seismic analysis approach used. Calculations were revised and confirmed to be acceptable.

Piping Analysis and Supports

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- The non-linear analysis using the time history method has been satisfactorily addressed by the applicant.
- The HRHF response spectrum seismic analysis of piping within the scope of the graded approach had been omitted. It has since been completed and will be included in a revision to technical report APR-E-S-NR-14004-P, which will be reviewed by the NRC staff.
- Requested from the applicant information regarding the structural integrity of piping and pipe supports that could be impacted by vibration and water hammer, which could potentially originate from the operation of the safety injection tank and its fluidic device. The applicant's response is pending.

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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

- Reviewed the applicant's criteria used to define the pipe break/leakage crack locations and configurations.
- Reviewed the outlines of the information which will be included in the pipe break hazards analysis (PRHA) summary report and requested the applicant submit a PRHA summary report.
- Reviewed the applicant's methodology for addressing the potential non-conservatism of ANS 58.2 dynamic jet modeling described in a technical report.

Staff Findings

- With the exception of break exclusion area, the applicant's criteria to define the pipe break/leakage crack locations and configurations are found acceptable.
- PRHA report outlines are found acceptable. The results of the PRHA summary report will be submitted in the future and remains open.

Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

- Technical Report (TR) methodologies in addressing jet plume expansion and distribution of the pressure within the jet plume are found acceptable. Evaluation of blast wave and potential feedback amplification and resonance effects remains open.

<u>Open Items Status</u>

- Break Exclusion Area and PRHA Summary Report

The design provisions to be employed in the break exclusion area are consistent with staff's guidance. However, the break exclusion areas are beyond the containment penetration area. The staff will review the applicant's justification with the results of pipe rupture analysis to be provided in the PRHA summary report.

- Blast Wave and Potential Feedback Amplification and Resonance Effects

The staff requested the applicant to clarify the CFD modeling for blast wave effects including V&V of computer code and the feedback amplification and resonance effects as presented in the current TR. Subsequently, the applicant indicated that the resulting dynamic load was too conservative in the current TR and, therefore, it will submit a TR revision with alternative approaches to address these issues.

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 mostly acceptable
- Most of the technical issues and response to RAIs by KHNP were acceptable and were therefore closed.
- One RAI remains as an open item.

Leak Before Break

- The staff questioned the PICEP input file for the surge line fluid temperature against what is provided in the DCD. In addition, the PICEP source code was requested if available.
- This RAI remains as an open item.
- Confirmatory analysis will continue upon receiving a response to the open item.

Special Topics for Mechanical Components
SER Section 3.9.1

Tom Scarbrough

Special Topics for Mechanical Components

- 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

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- 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

- 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 proposed to track design specification and DCD changes.

- Open item to be closed when RAI resolved.

Functional Design, Qualification, and IST Program
SER Section 3.9.6

Tom Scarbrough

Functional Design, Qualification, and IST Program

- 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 will be reviewed to close Confirmatory Items.

Functional Design, Qualification, and IST Program

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- Open Item: Design Specification Follow-up Audit items for functional design and qualification are acceptable. Follow-up Audit Report being prepared.
- Open Item: Proposed IST table revision in August 29, 2016, submittal is acceptable and will be confirmed by review of DCD Revision 1.
- Open Item: KHNP preparing responses to RAIs on pump and valve ITAAC.

Environmental Qualification of Mechanical and Electrical Equipment SER Section 3.11

Jorge A Cintron

Environmental Qualification of Mechanical and Electrical Equipment

- 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

Open Item

- Conformance with **Regulatory Guide (RG) 1.89:** provides the acceptable method for environmental qualification. Endorses IEEE 323-1974, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations."
- The APR1400 deviates from RG 1.89 by using IEEE 323-2003, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," for environmental qualification of mechanical and electrical equipment.
- Staff recognize the use of standards not endorsed by the regulatory guides if it is appropriately justified.
- The staff issued an RAI requesting justification why IEEE Std. 323-2003 is acceptable for qualification of Class 1E electrical equipment in the harsh environment

Environmental Qualification of Mechanical and Electrical Equipment

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<u>Open Item (Cont.)</u>

- In response to the RAI, the applicant determined to use IEEE 323-2003 for environmental qualification stating that there are not technical differences, and it the reflects current practices for environmental qualifications.
- The staff identified technical differences in the content of IEEE 323-2003 and issued a follow-up RAI requesting the applicant to provide justification of the technical difference between IEEE 323-1974 and IEEE 323-2003.
- Staff is currently evaluating the applicant revised response to the above RAI, which is an Open Item for Section 3.11

Radiological Equipment Qualification Review SER Section 3.11

Ed Stutzcage

Equipment Qualification (Radiological)

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- Staff reviewed the applicant's methodology and results for calculating the total integrated dose (TID) to equipment within the EQ program.
- Normal operation doses for equipment are based on the highest dose within a room, using a similar methodology to that used to determine the Chapter 12 radiation zone maps, except based on an assumed 1% failed fuel percentage (instead of 0.25% assumed in Chapter 12), and a few other differences.
- Accident doses are based on the most limiting design basis accident for each area of the plant (for most areas the design basis LOCA is limiting).
- TIDs are the sum of the 60 year normal operation dose, plus the limiting accident doses.

Results and Conclusions (Equipment Qualification Radiological)

- For most areas of the plant, staff finds the applicant's normal operation dose values for equipment qualification to be acceptable.
- Outstanding issues include:
  - Inconsistencies with Chapter 12 information regarding normal operation neutron dose on the refueling floor.
  - Doses within the Auxiliary Building during accidents do not appear to adequately consider radiation streaming through containment penetrations.
  - Requested the applicant to provide additional information regarding how some of the post-accident gamma dose rate information was determined.
  - Unjustified assumptions for post-accident fluid leakage rate outside of containment.

ACRONYMS

- 3-D – three dimensional
- AB – auxiliary building
- CIS – containment Internal structures
- COF – coefficient of friction
- COL – combined license application
- CSDRS- certified seismic design response spectra
- DFOT – diesel fuel oil tank
- EDGB – emergency diesel generator building
- FEA – finite element analysis
- HRHF – hard rock high frequency
- IRWST – in-containment refueling water storage tank
- ISRS – in structure response spectra
- NI – nuclear island
- PSD – power spectral density
- RAI – request for additional information
- RG – regulatory guide
- RCB – reactor containment building
- SRP – standard review plan
- SSI – soil-structure interaction
- SSSI – structure-soil-structure interaction
- SSW – secondary Shield Wall
- UPC – ultimate pressure capacity