Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title:	Advisory Committee on Reactor Safeguards Open Session
Docket Number:	(n/a)
Location:	Rockville, Maryland
Date:	Wednesday, July 12, 2017

Work Order No.: NRC-3164 Pages 1-117

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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645TH MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS) OPEN SESSION

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WEDNESDAY

JULY 12, 2017

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ROCKVILLE, MARYLAND

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The Advisory Committee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B1, 11545 Rockville Pike, at 8:30 a.m., Dennis C. Bley, Chairman, presiding.

COMMITTEE MEMBERS:

DENNIS C. BLEY, Chairman
MICHAEL L. CORRADINI, Vice Chairman
PETER RICCARDELLA, Member-at-Large
RONALD G. BALLINGER, Member
CHARLES H. BROWN, JR. Member
MARGARET CHU, Member
WALTER KIRCHNER, Member
JOSE MARCH-LEUBA, Member
DANA A. POWERS, Member
GORDON R. SKILLMAN, Member
JOHN W. STETKAR, Member

DESIGNATED FEDERAL OFFICIAL:

CHRISTOPHER BROWN
KENT HOWARD

ALSO PRESENT:

TONY AHN, KHNP

ARDEN ALDRIDGE, STPNOC

DENNIS ANDRUKAT, NRO

JOE ASHCRAFT, DEI

THOMAS BERGMAN, NuScale

ARACELI BILLOCH, DLR

THERESA BUCHANAN, NRC

ANGELA BUFORD, NRR

ALEX BURJA, NRO*

EDWARD CARLEY, NextEra Seabrook

MARK CARUSO, NRO/DSRA/SPRA

NAN CHIEN, NRO

WOOCHONG CHON, KEPCO E&C

JORGE CINTRON, NRR

RUSSELL CIPOLLA, Intertek

PHYLLIS CLARK, DLR

STEPHEN CUMBLIDGE, NRR

JOHN DAILY, Member of the public

ROB ENGEN, STPNOC

ROBERT FITZPATRICK, NRR

ADAKOU FOLI, NRR

BART FU, DLR

ISMAEL GARCIA, DEI

MICHAEL H. GARNER, STPNOC

RON GIBBS, STPNOC

JAMES GILMER, NRO

VIJAY GOEL, NRC

RAFAEL GONZALES, STPNOC

NICHOLAS HANSING, NRO

RAUL HERNANDEZ, NRO

ALLEN HISER, DLR

WILLIAM HOLSTON, DLR

JOHN HONCHARIK, NRO

ATA ISTAR, NRO

DIANE JACKSON, DSRA

LOIS JAMES, DLR

JAE HOON JEONG, KEPCO/NF

JONG HODU JHEOU, KEPCO E&C

REBECCA KARAS, NRO

RAIHAN KHONAKER, STPNOC

HANGON KIM, KHNP

JUNGHO KIM, KHNP

YOUNGKI KIM, KEPCO E&C

MATTHEW KING, STPNOC

REBECCA KARAS, NRO

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JOHN MA, NRO
GREG MAKAR, NRO
TANIA MARTINEZ NAVEDO, NRC
MIKE MCCOPPIN, NRO
SEUNG MIN, DLR
AARON MINK, DLR
JILL MONAHAN, Westinghouse
MICHAEL MURRAY, STPNOC
JINSUO NIE, NRO
CAROL NOVE, RES
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GREGORY PICK, Region IV*
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ERIC REICHELT, NRO
DAVE RENCURREL, STPNOC
BILL ROGERS, DLR
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MOHAMMAD SADOLLAH, NRR

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GEORGE WILSON, DLR
BRIAN WITTICK, NRR
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GEORGE WUNDER, NRO
ZUHAN XI, NRO
MARK YOO, DLR
JINKYOO YOON, KHNP and KEPCO E&C
CRAIG YOUNGER, STPNOC

*Present via telephone

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P R O C E E D I N G S

8:30 a.m.

CHAIRMAN BLEY: This is the 645th Meeting of the Advisory Committee on Reactor Safeguards. During today's meeting, the committee will consider the following: 1) License Renewal Application for South Texas Project; 2) NuScale Topical Report on Safety Classification of Passive Nuclear Power Plant Electrical Systems; 3) Advanced Power Reactor 1400; 4) WCAP on Westinghouse Performance Analysis and Design Model L for PAD5; and 5) preparation of a ACRS Reports.

The ACRS was established by statute and is governed by the Federal Advisory Committee Act. As such, this meeting is being conducted in accordance with the provisions of 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 Federal Register Notice describing a meeting as published.

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

Written comments are also welcome. Mr. Kent Howard is the designated federal official for the initial portion of this meeting. I knew I saw you earlier.

Portions of the sessions on NuScale Topical Report and the APR1400 may be closed in order to discuss and protect information designated as proprietary.

The WCAP meeting will be closed in its entirety in order to discuss protective proprietary information.

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

We have received no written comments or requests to make oral statements from members of the public regarding today's session. There will be a phone bridge line. To preclude interruption of the meeting, the phone will be placed in the listen-in mode during the presentations and committee discussion.

A transcript of portions of the meeting is being kept and it is requested that the speakers use one of the microphones, identify themselves, and speak with sufficient clarity and volume so that they can be readily heard.

At this time, we turn to the South Texas Project License Renewal Application and I turn the meeting over to Dick Skillman. Dick?

MEMBER SKILLMAN: Dennis, thank you.

Good morning. This is Agenda Item 2, License Renewal Application for the South Texas Project. This morning, we will hear presentations from the Division of License Renewal and the applicant, South Texas Nuclear Operating Company.

Our License Renewal Subcommittee previously met to discuss this matter on November 17, 2016. At the conclusion of that meeting, there was one open item and that item had to do with the selective leaching of aluminum bronze.

Today, we will hear from the applicant on addressing enclosure of this item and from the staff. And with that, I call upon and welcome George Wilson to begin the presentation.

George.

MR. WILSON: Thank you, Chairman Bley, Mr. Skillman, and members of the ACRS. I am George Wilson, the Director of the Division of License Renewal. With me at the table is Sheldon Stuchell, the Branch Chief for the Projects Branch.

Presenting for the NRC today will be Lois James, the Senior Project Manager for South Texas, Bill Holston, the Senior Mechanical Engineer, and Dr. Allen Hiser, the Senior Level Advisor for License Renewal.

Also behind me in the audience are members of technical review team. We look forward to a productive discussion today while presenting our safety evaluation report for the South Texas Project Units 1 and 2, which determined that the requirements of 10 CFR 54.29(a) have been met for the license renewal of South Texas Project Units 1 and 2.

During the subcommittee, we discussed one open item associated with the Selective Leaching of Aluminum Bronze Aging Management Program. This item is now closed and the staff will be discussing its resolution during our presentation.

At this time, I would like to turn the presentation over to the South Texas Project Nuclear Operating Company and Dave Rencurrel, Senior Vice President of Operations, to introduce his team and commence their presentation.

MR. RENCURREL: Thank you, George.

Good morning. My name is Dave Rencurrel.

Good morning, Mr. Chairman. Before we get started, I would really like to thank George and his staff and, specifically, Lois, our Project Manager, for the hard work and the preparation of the SER and really accepting of our application.

And I would also like to thank you and the ACRS for this opportunity to present our License Renewal Application. I look forward to the questioning.

Our next slide, this shows our agenda. And as you can see, we are going to present some background information about our station and then a high-level discussion of our application. And we do have an agenda item to focus on the open item that we had for selective leaching.

It was our opinion, our working, we really feel that we believe that we have developed a robust, high-quality License Renewal Application, that our Aging Management Programs provide the reasonable assurance for the continued safe and reliable operation of our station and we appreciate this opportunity today to make this presentation. I look forward to answering any questions that you all may have.

Let's start with some introductions. As

said earlier, my name is Dave Rencurrel. I am the Senior Vice President of Operations. My current position at the staff after 29 years is I am responsible for all the major projects on-site, all the construction activities, and then the contracts associated with those. Part of those projects is the license renewal project.

Throughout my history, I have also had very many jobs. I started off in the Navy in 1981 and served seven years as the nuclear officer and I jointed South Texas in 1988 and I have been at South Texas since '88. So, it is really over 29 years now. I have had many jobs, like I have said, up to and including site VP and in my current role.

And to my left I have Michael Murray.

MR. MURRAY: Good morning. Michael Murray. I have been in the industry for 42 years. Don't try to do the math on that one. But at STP for 32 years, I was there for startup of both units. So, I had the opportunity of actually working in the startup of the units.

At South Texas Project, I have had various opportunities in supervisory and management positions. Most recently, System Engineering Manager. Then I worked as I&C Design Manager for the Units 3 and 4 licensing activities. And currently, I am the Regulatory Affairs Manager.

MR. GIBBS: Good morning, I'm Ron Gibbs. I have over 31 years in the nuclear industry. I started in 1986 up at the Comanche Peak Nuclear Power Plant. I received an SRO in 1988 and was Unit Sup STA during initial startup and testing.

I joined the South Texas Project in 1993, obtaining my senior reactor operator license in 1995 at the South Texas Project. I was a member of the on-shift control room for 17 years as a Unit Sup STA and the progressing up through a shift manager.

In 2013, I joined the Operations Management Team. I was the senior license at the South Texas Project for about two years and transitioned in April to my current position, which is Ops Manager in Training.

MR. ALDRIDGE: Good morning. My name is Arden Aldridge. I've been in the South Texas Project 25 years in various engineering roles. Currently, the License Renewal Application Project Lead and will be the License or am the License Renewal Implementation Coordinator.

MR. RENCURREL: All right, thank you Arden. And throughout the room we have our Aging

Management Plan subject matter experts. We have folks from both design engineering and programs engineering and we also have some specialty consultants here to help support and help answer any questions that you all may have.

Let's give an overview right quick of our station ownership and operation. The South Texas Project Nuclear Operating Company is a company that parks on top of the asset. We don't own the asset. We operate the asset and maintain the asset for three owners. And our license is assigned to the South Texas Project Nuclear Operating Company. Our three owners are NRG Texas, the City Public Service Board of San Antonio or CPS Energy, and the City of Austin or Austin Energy.

I can say that our owners are committed to providing the right level of resource support and oversight to ensure that we have safe and reliable operation at South Texas and that we maintain and that we stay a critical long-term power generator for the State of Texas.

This slide shows a list of some of the major capital improvements that have been made throughout the history of South Texas. As you can see by the slide, our initial license was granted in 1987 and 1988, respectively, for Unit 1 and 2. Since that time, our owners have invested heavily to position our station for long-term, safe, reliable operation. You can see that we have changed out our steam generators. We have updated our low pressure turbines. We've replaced our reactor vessel heads. For the main generators, the main electrical generators, we both replaced the stators and the rotors. We completed our Alloy 600 program with the mechanical stress improvement process -- complete might not be the right word but feel committed to exercising the Alloy 600 program.

And also what's not shown here is we also have changed out our main transformers and are in the process of changing out our large feedwater heaters. And so we are positioning the unit for long-term reliable operation.

I can also say that in regards to the Aging Management Plan and the commitments in the Aging Management Plan, the governance that we use is that we have what is called a Plant Investment Plan which is incorporated into our business plan. What that Plant Investment Plan does is it ensures that we have the right funding and the right resources committed to ensure that the commitments of the Aging Management

Plan are carried out appropriately and they are carried out in a way that they will be completed well before the expiration of the current operating period.

That concludes my brief introduction. And with that, I would like to transfer over to Ron Gibbs, our Ops Manager.

MR. GIBBS: Good morning, again, Ron Gibbs.

So I would like to give a site description here. The South Texas Project is located near this star, as annotated over here in the State of Texas. We see it by the aerial view. We are in a rural area in Matagorda County, which is about 90 miles southwest of Houston and we are about 15 miles inland from the Gulf.

The big area in the middle most notable is our 7,000-acre main cooling reservoir, which we sit on a 12,000-acre site. The main cooling reservoir is made up from the Colorado River to the east of the plant here and from rainwater.

On the station description here you can see the main cooling reservoir, which I was just explaining on the top here.

At the bottom here is our central cooling water pond. That's commonly referred to in the industry as service water and that is our ultimate heat sink. Makeup to our ultimate heat sink is from well water and rainwater.

On the right here, you can see our switchyard. And in the center are the two units of South Texas Project. They are separate units, are Westinghouse four-loop pressurized water reactors, 3853 megawatts thermal and 1250 megawatts electric for both units.

Operators are licensed on both units and we use common operating procedures.

Containment structures are semispherical heads with steel liners and flat bottoms. Each unit has three independent safety-related trains. That includes all our pumps, piping, valves, and diesel generators, and each unit has four auxiliary feedwater trains; three electric-driven pumps and one steam-driven pump for each unit.

Next, I would like to turn it over to Arden to give a status of our license application.

MR. ALDRIDGE: Good morning; Arden Aldridge, License Renewal.

The slide here just kind of gives us a historical perspective of where we've been. We submitted our application in October of 2010 and during that time, we have completed all the different reviews, annual updates, integration of operating experience through the Interim Staff Guidances and responses to Requests for Additional Information from the NRC staff.

And here we are today. So, we're excited to be able to present this.

The GALL Consistency Table is the same as what you saw during the subcommittee meeting. I think there might be a few little changes with the things that we incorporated since then. But here, it basically tells us that we have 41 Aging Management Programs; 8 new ones and 33 existing programs of which we have various enhancements and exceptions and plant-specific.

The STP License Renewal Application addressed all the lessons learned identified in GALL Rev. 2 and all the other requirements of the standard review plan.

Out of that review and through that process of the license renewal, we established 47 commitments and those commitments are included in the SAR supplement, which is Appendix A of the License Renewal Application and will be managed through the South Texas Licensing Commitment Management administrative processes.

Eight of those commitments have already been implemented and 39 commitments remain to be completed. To give you a perspective, those commitments represent 22 procedural enhancements or new procedures, 13 inspection scopes, 2 replacement scopes, and 2 additional analysis updates specifically around fatigue cycle counting and 62060 sentinel locations.

We have an implementation plan, a schedule and budget to complete all the remaining commitments on their scheduled due dates, prior to entering the period of extended operation in 2027 and 2028.

So, why are we here today? To wrap up the things that have happened since the subcommittee meeting. We have three activities that were incorporated. The first one was we clarified the selective leaching of Aluminum Bronze Management Program to close this open item. And we did that using -- we identified and optimized the use of a non-destructive examination to manage the age-related degradation of the aluminum bronze weld materials.

The second one is we revised the Steam Generator Aging Management Program to incorporate lessons learned from interim staff guidance 2016-01 and in that, we added additional steam generator component scope inspections to the program.

And then the third one is we updated the Bolting Integrity and External Surfaces Monitoring Aging Management Program to add some additional clarification for inspection methods to be used to detect leakage associated with closure bolting in air-filled and gas-filled systems.

So the last two were both incorporation of lessons learned during the -- from the time of the subcommittee to today.

MEMBER STETKAR: Arden, on the selective leaching, I lose track of meetings. I read the revised app and what you are going to do about the welds. Didn't you also commit to replacing all aluminum bronze castings? I don't know what the -- it must be to help valve bodies and things like that. Was that done at the time -- was that done before our subcommittee meeting or was that also done during this interim period?

MR. ALDRIDGE: No, that was addressed as part of the attributes of the Aging Management Program that we've developed. That is part of the Aging Management Program.

MEMBER STETKAR: Are we talking about

primarily valve bodies or are they heat exchanger components or what?

MR. ALDRIDGE: To answer that, primarily we are talking about flanges.

MEMBER STETKAR: Oh, flanges.

MR. ALDRIDGE: Flanges but there are valve bodies and a couple of T's and then the pump casings have aluminum bronze.

MEMBER STETKAR: Have you got underground valve bodies?

MR. ALDRIDGE: No, we do not.

MEMBER STETKAR: Not in the central area.

MR. ALDRIDGE: That's a separate one.

MEMBER STETKAR: Okay, thank you.

MR. ALDRIDGE: That concludes my presentation.

MR. RENCURREL: And that concludes our opening comments, our remarks. Again, we are committed to the safe, long-term, reliable operation of South Texas. As you can see by introductions here, many of us have really dedicated our lives and our efforts to make sure this station is successful and this is a big part of what we wanted to do. So, we thank you very much for this opportunity.

MEMBER SKILLMAN: Dave, thank you.

MEMBER STETKAR: I do have a follow-up question because I'm not a materials person. And I'm sorry to come in after the fact a bit.

I noticed that the AMP does -- I don't know whether it is all or most of -- the inspections on the above-ground sections of piping on the welds. Can you give us your rationale about why you feel that those inspections provide adequate information to understand what's going on on the below-ground welds?

I understand you are going to do episodic examination -- or I don't want to say examinations. You are going to check things if you have to ever excavate the pipe but that is on an opportunistic basis. Your program is primarily organized around the readily accessible above-ground piping.

So do why do you feel that that gives you confidence in the status of the below-ground welds?

MR. ALDRIDGE: Alden Aldridge responding. The difference between the above-ground and below-ground, as we established, the below ground components, as far as the piping and the valves are not susceptible. They are out of rock material that don't dealloy like the castings and stuff do. That addresses the overall system.

To specifically address your question on welds, which the welds are the same, whether they are below ground or above ground, is we are doing a sampling of all the above-ground welds, which represent the total population. And based on the as-found conditions, the initial criteria is the weld material itself is not -- it's susceptible but we have had no operating experience of significant dealloying in the welds at the station after the initial startup. So, we have that to start with.

The second thing that we have is the margin of between the stress requirements of the below-ground and above-ground, we have about a two-times margin of available stress margins between the above-ground and the below-ground. So if we find anything on the above-ground, we have the opportunity to expand the scope, which is part of the Aging Management Program, to below-ground scopes.

So, the program doesn't ignore them but it uses the above-ground as the leading indicators and we have sufficient time to be able to do a recovery.

MEMBER STETKAR: That's what I was looking for is why do you have confidence that those above-ground welds are appropriate as leading indicators.

MR. ALDRIDGE: Right.

MEMBER RICCARDELLA: So would the sampled expansion automatically go to some underground welds if you found unacceptable indications?

MR. ALDRIDGE: It depends what we find. It doesn't necessarily immediately go to it but if we have -- we would go to stress margins, stress locations where we had the lowest margins.

MEMBER SKILLMAN: Dave and team, thank you. At this point, let's change out and go to the NRC team, please.

Lois, welcome. Let's just give a minute here so we can get the bridge line open. Bill, Allen, welcome.

We are holding here to make sure we've got Greg Pick on the line. He is one of the inspectors.

MEMBER STETKAR: Just ask him if he's there.

CHAIRMAN BLEY: Greg, are you on the line? Can you speak?

MR. PICK: I am, sir.

CHAIRMAN BLEY: Good.

MEMBER SKILLMAN: There we go. Greg, welcome and thank you.

Okay, Lois, back to you. Please take the lead.

MS. JAMES: Okay. Good morning, Chairman

Bley, Mr. Skillman --

MEMBER SKILLMAN: You need a green light. There you go.

MS. JAMES: Usually my mouth voice is enough.

Good morning, Chairman Bley, Mr. Skillman, and members of the ACRS. My name is Lois James and I am the License Renewal Project Manager for the South Texas License Renewal Safety Review. We are here today to discuss the South Texas Project License Renewal Safety Evaluation Report, which was issued just a month ago.

Joining me here at the table are Dr. Allen Hiser, the License Renewal Senior Level Advisor; Mr. William Holston, Senior Mechanical Engineer; and Ms. Phyllis Clark, Project Manager, who will run the slides.

Joining us by phone is Mr. Gregory Pick from Region IV, who can answer any inspection-related questions.

Seated in the audience and joining by phone are other members of the technical staff who participated in the review and the audits. Next slide, please.

I'll begin the presentation with a general overview of the staff's review. Mr. Holston will discuss the closure of the open item regarding the Selective Leaching of Aluminum Bronze Aging Management Program and then I'll present the staff's conclusion. Next slide.

South Texas Project Nuclear Operating Company submitted its application for South Texas Projects Unit 1 and 2 in October of 2010. The staff issued two Safety Evaluation Reports with Open Items, one in February 2013 and one in October of 2016.

We presented the 20167 SER with Open Items to the ACRS subcommittee in November.

The staff closed the final open item and issued the final SER on June 8th. Next slide.

The applicant identified four Aging Management Programs in its application and subsequently added an additional existing program in response to RAIs issued by the staff, based on interim staff guidance. The left side of this slide identifies the applicant's disposition of its AMPs. The right side identifies the staff's disposition in the final SER issued in June.

All plans were evaluated by the staff for consistency with the GALL Report or the Standard Review Plan, as appropriate. The applicant may enhance existing programs to be consistent with the programs, as described in the GALL, or they may take exceptions to these programs.

Throughout the staff's review, AMPs were updated and revised based on RAI responses on the application, as amended; changes to the plant, as documented in the annual updates; and RAI responses, based on generic issues identified by the staff during reviews of other license applications.

Since the subcommittee meeting in November of 2016, several AMPs have changed disposition categories. I will now turn the presentation over to Mr. Holston to discuss closure of the open items.

MR. HOLSTON: Good morning. The staff has closed the open item associated with the Selective Leaching of Aluminum Bronze Aging Management Program.

CHAIRMAN BLEY: Mr. Holston?

MR. HOLSTON: Yes, sir?

CHAIRMAN BLEY: As you go through, I would appreciate it -- I wasn't at the subcommittee meeting -- if you could give us a little background on how this actually arose.

MR. HOLSTON: Yes, sir, I will. That's on the next two slides.

So during the ACRS subcommittee meeting, the staff discussed the closure of many aspects of the open item. And those examples of those were the extent of the destructive examinations that were going to be conducted, acceptance criteria, et cetera. However, the remaining open portion of that open item was that the AMP did not adequately address corrective actions associated with inspection results, demonstrating that structural integrity requirements would not be met.

As you requested, I will provide an over of the scope of the program for those members that are not on subcommittee and then discuss the proposed additional inspections, acceptance criteria, and corrective actions associated with closure of the open item. Next slide, please.

MEMBER SKILLMAN: Bill, before you proceed, let me go back to Lois' slide 3, please. Just checking my notes against my notes. I see that we met at 0830 on November 17th, not November 18th. It's a nit detail but, for the record, that is the date that we met, November 17, 2016.

MS. JAMES: Yes.

MEMBER SKILLMAN: Okay?

MS. JAMES: Okay.

MEMBER SKILLMAN: Excuse me, Bill.

MR. HOLSTON: No problem.

MEMBER SKILLMAN: Please proceed.

So the applicant developed a plant-specific Aging Management Program to address selective leaching of aluminum bronze and its essential cooling water system. That report has an existing Aging Management Program, AMP 33 for Selective Leaching; however, the purpose of that program is to determine whether you are experiencing selective leaching at the plant and it addresses more than just aluminum bronze; it addresses copper, alloys greater than 15 percent zinc; it addresses cast iron or gray cast iron.

The applicant, recognizing that it had specific operating experience related to selective leaching of aluminum bronze, developed this plant-specific Aging Management Program.

Loss of material due to selective leaching of aluminum bronze principally occurs if the aluminum content is at greater than eight percent and if there is a slow cool down rate of the casting or the weld. And what occurs, if the cool down rate is slow enough, your form susceptible beta and gamma-2 phases in the lattice structure and the alpha phase is not susceptible. So that's what occurs. These welds and/or castings had beta and gamma-2 phases that were susceptible.

So at South Texas, they have approximately 350 remaining castings that are susceptible. They have susceptible welds due to the filler metal aluminum content and that is approximately 3400. And the piping material, as the applicant already described, is not susceptible; that aluminum content is low enough.

MEMBER POWERS: Is that a rock piping material?

MR. HOLSTON: Say again?

MEMBER POWERS: Is that a rock piping material?

MR. HOLSTON: Yes, sir. Next slide.

So, since 1987 and through 2015, 55 through-wall casting defects have occurred as a result of loss of material due to selective leaching. The applicant has developed an existing program that they have been implementing for the current period of operation that does routine visual examinations, that they detect indications of leakage. They have an extrapolation where they say you know if we see a little line about this long along a casting, then they extrapolate the volume of the internal selective leaching. That extrapolation was based upon six samples that they conducted, where they destructively examined it and actually measured the extent of the dealloying.

They then put that volume into a structural integrity evaluation, determined if it meets structural integrity requirements. If it does, they submit for code relief because they have a leak, so it doesn't meet Section 11 Class 3 components. And the NRC has typically accepted those code release and granite relief and then they replace the component at the next refueling outage. So, that's what they have been doing with castings.

And they have had no leaks since --

MEMBER SKILLMAN: Excuse me, Bill.

MR. HOLSTON: Yes, sir.

MEMBER SKILLMAN: For those incidences where there is a crack or an indication, what is the approximate leak rate, drops per minute, drops per hour, gallons per minute?

MR. HOLSTON: At least for all the very recent ones. I'm not familiar with back in the '80s. You may not even measure any leakage at all. For instance, the one I saw, they only had one leaker when I was there on-site. I have been on-site a few times and that was just you saw a light green kind of florescent haze and there wasn't any water on the floor.

But there have been leakers. They have addressed that with leak-limiting devices, where necessary. And that's all addressed in their UFSAR as part of their current licensing basis.

MEMBER SKILLMAN: Thank you.

MEMBER BALLINGER: Selective leaching results in -- what you're saying, the strength of the material is not so affected but the impact strength is affected.

So what happens if you have a region that has got a lot of selective leaching? So you have got this sort of semi-porous, if you want to call it, material and then, for lack of a better word, somebody whacks on it, or hits it, or something like that. Have you thought about what happens when that occurs or if that were to occur?

MR. HOLSTON: Yes. When the calculations that they do -- they do a linear elastic-plastic fracture mechanics calculation -- they remove all credit for the dealloyed regions. So they are only crediting the un-dealloyed portion of the remaining fit.

So for example, the most severely affected fitting that we had seen of the results they had had 60 percent average dealloying around the entire fitting. And so that 60 percent material was, in effect, gone from the analysis. And when the plug it in, it is only the remaining 40 percent that they credit.

MEMBER BALLINGER: But if it's a leak, by definition, that means you have got a through-wall --

MR. HOLSTON: Yes.

MEMBER BALLINGER: -- thing. And so what about impact loading on that? Because now it's not a fracture mechanics problem. You have no new material, no unaffected material to deal with.

MR. HOLSTON: Well, I mean that's all factored into the calculations.

MEMBER BALLINGER: Okay.

MR. HOLSTON: So in other words when I said 60 percent average, I was just trying to describe the degree of degradation. They actually model it with where you know like here it is through-wall. Here it is about 30 percent you know take readings around about every 12 and a half degrees, if I recall right, and then do a profile of the actual degradation.

So you're right. There is some portion

that goes completely through-wall. That is all factored into the calculation.

DR. HISER: Yes, this is Allen Hiser and I think the geometry that they use is consistent, as if it were cracked.

MEMBER BALLINGER: Okay.

DR. HISER: So if there is remaining material and it will sustain load. So, it is not as if the entire cross-section is dealloyed.

MEMBER BALLINGER: Yes, I'm sure that it will sustain the load but I have a 180-year-old house where I replaced the cast iron pipe. All you needed to do was to hit the elbow with a hammer and it shattered because of the selective leaching of the gray cast iron.

DR. HISER: And my guess is with this, if you had a sufficient extent of the alloy, you would have the same effect. But I think the fracture analyses that they have performed are intended to safeguard against any operational or accident loads. So, the analysis should take that into account.

MEMBER BALLINGER: Including impact load.

DR. HISER: Whatever the design-basis loads are. I can't speak specifically what impact loads are included in their design basis.

MEMBER BALLINGER: Because I couldn't find that in the reading I did.

MEMBER SKILLMAN: Well for a minute there, I thought you were talking about subsequent life renewal at 180 years. That was intended to be a joke. I'm sorry.

Let's keep on going.

MEMBER RICCARDELLA: Excuse me, Bill, you used the term linear elastic-plastic analysis. That's the term, linear elastic-plastic. It is either linear elastic- or non-linear elastic-plastic, right?

MR. HOLSTON: You are correct, yes. Sorry.

MEMBER RICCARDELLA: Which was it?

MR. HOLSTON: Give me a minute here. I can't remember. It's been too long since I've looked at those calculations because they were all related to the casting.

MR. CIPOLLA: My name is Russ Cipolla and I am the one that has done these calculations.

The calculations are done in two parts. There is the linear elastic part to deal with any kind of low toughness material part of the alloy and then there is the limit load. So we do both and then we take the minimum of the two calculations to keep it simple to address both the behaviors.

CHAIRMAN BLEY: Would you also state your organization, for the record?

MR. CIPOLLA: I'm with Intertek. I am a contractor to South Texas Project.

MEMBER BALLINGER: But which one of those deals with impact analysis?

MR. CIPOLLA: Well, again, to deal with the impact, first of all, there is really no impact loads in the system. But we, as was being explained, we take no credit for the dealloyed material. So, if we have something that is leaking, whether it is a visible crack or whether it is just dealloyed, we take no credit for that. We assume a through-wall crack, analytical crack, and that's looking into virgin properties. So the virgin properties are very, very doubtful.

MEMBER BALLINGER: So you have no forklifts on the site?

MR. CIPOLLA: I can't answer that. I'll let South Texas answer that question.

MEMBER STETKAR: Or it seems like it would affect the seismic analysis, right?

MEMBER BALLINGER: I was going to say that, too, but I wasn't sure.

MR. HOLSTON: So, have we satisfied your question or do you have some follow-up there?

MEMBER BALLINGER: I still have some questions about the impact part.

MR. HOLSTON: Okay. Should we proceed and then see if we go back to that if the rest of this doesn't -- yes, okay.

MEMBER STETKAR: Bill, let me because I didn't -- in your SER, you refer to a letter where they made this commitment. And I didn't try to find the letter or the commitment to replace the castings.

In the SER it says they are going to replace all -- the word "all" is here -- aluminum bronze casting susceptible to selective leaching. When are they going to do that, before the extended period of operation or when?

MR. HOLSTON: Yes, they are going to replace all those castings prior to a period of extended operation.

MEMBER STETKAR: Thank you.

MR. HOLSTON: Okay, do you have questions before I go on?

Okay, so we talked about castings. Just to bring us back to where we were in the first bullet. There has been 55 through-all casting defects occurring from '87 to 2015.

And since 1989 and progressing through 1994, there were seven leaks in welds with backing rings. These leaks originated from weld defects that, in most cases, progressed in part due to selective leaching.

And based on testing by the applicant, the root pass of the weld is less susceptible to loss of material due to selective leaching because it has a faster cool down rate and there is a lower aluminum content.

The faster cool down rate was based upon calculations that South Texas ran and then we, in Research, we had an individual who did independent calculations and confirmed that the cool down rates are faster in the root pass than in subsequent passes.

And lower aluminum content was determined from based upon destructive examinations that they took apart six welds and characterized those. And if you'd like to talk about that a little bit further, we have some backup slides for that.

So based upon that, if the root pass is less susceptible, as long as you can retain the root pass intact, then it is unlikely that you will have selective leaching of aluminum bronze through the entire weld.

So based upon that, the applicant significantly revised their AMP in 2016. As we were just discussing, all susceptible cast components will be replaced and the susceptible welds joining non-susceptible piping components will not be replaced. And as I said, there is approximately 3400 of those.

MEMBER STETKAR: Bill?

MR. HOLSTON: Yes, sir.

MEMBER STETKAR: I know nothing about materials. So, bear with me here.

Your first bullet here says apparently at least through 2015, they were still experiencing through-wall defects in cast components. But for some reason, they haven't seen any weld problems in the last 23 years. Is there a physical reason for that?

I mean why? Do you understand?

MR. HOLSTON: No, no, I do understand. The welds that did fail all had construction defects in the welds and that penetrated through the root pass. And so that allowed -- now, with the weld defect, that allowed the environment, the central cooling water, to progress. And then what they found in most of those welds is kind of a successive you had the defect basically crack and then it dealloyed some;

the crack continued; it dealloyed some and the crack continued.

MEMBER STETKAR: What I was looking for is is there, because of that operating experience during the five years that you've listed there or up through 1994, is that added confidence that they have essentially found the welds that had initial defects and that --

MR. HOLSTON: We believe that yes, the welds found them. Now, they did some extensive volumetric examinations. I mean they knew the right thing to do back in 1990; let's go look at a lot more welds.

The program going forward does some sampling of welds to account for that. So we're not just out measuring the dealloyed.

MEMBER STETKAR: That's right.

MR. HOLSTON: Yes.

MEMBER STETKAR: I'm just trying to look at confidence-building; that we're not in some sort of bath tub curbed region or something like that.

MR. HOLSTON: Correct. We're confident but we're going to make sure the AMP has the provisions to beef up that confidence.

MEMBER STETKAR: Okay, thank you.

DR. HISER: Confidence but not assurance from that operating history.

MEMBER STETKAR: Okay.

MR. HOLSTON: So are there any more questions on the background? Because my next slides go into actually the closure. Do you all feel you have a good feel for what's going on with the plant?

Okay, I want to take a brief moment to talk about a UT method that was developed by South Texas working with EPRI to detect an extended dealloying. And this method was essential to developing appropriate corrective actions that gave us assurance that the system would meet its intended functions.

I could give you an overview. If you have some detailed questions, I have the read heavy lifters in the room and that's Stephen Cumblidge and Carol Nove in the back, if you want to get into the very deep details of this technique but I can talk in general terms.

So, this time of flight diffraction UT method, it isn't the first time it's invented; it's been used in many, many applications all across the world. It will detect circumferential extent and the depth of the dealloyed material. So basically, you take the weld; you do the time of flight; and it can tell you how much circumferentially around the weld and the depth of it so you get the total volume of the selective leaching within the lattice structure.

The South Texas Plant developed a plant-specific procedure. It was developed in accordance with ASME Section V for nondestructive examination and the staff reviewed the TOFD, time of flight to UT method. They looked that the validation tests. They looked at the implementing procedures. And they looked at the personnel requirements and found them acceptable to detect dealloying.

We did not validate this method for all across the industry. We looked at their specific plant procedures and their specific testing. We are confident that it will detect the volume of dealloying but we did want to tell ACRS that if we were to accept this method across all sorts of applications, we would do deeper research.

Any questions on time of flight method?

MEMBER RICCARDELLA: Was this done along the lines of the EPRI PDI program or was there applied samples?

MR. HOLSTON: It was not the EPRI program. They used the low rigor method in Section V. And so basically, they developed a technical basis document. They did some -- they did six samples but they weren't necessarily blind samples and so that met ASME Section V for low rigor. We accepted low rigor because these are not ASME Section XI inspections. They are to determine extent of condition of the dealloying if they come up with adverse results that don't meet acceptance criteria.

MEMBER STETKAR: But Bill, I think you just said it. In their program, this is not the primary method that they use for examination of the welds. This is something that they will employ only if they had indications of degradation. Is that correct?

MR. HOLSTON: That's correct.

MEMBER STETKAR: Okay.

MR. HOLSTON: Yes, and that will actually be in the next slide. We have a flow chart there.

MEMBER STETKAR: Oh, sorry.

MR. HOLSTON: But no, no, that's exactly correct. Yes, sir.

CHAIRMAN BLEY: You've got a lot of straight men.

MEMBER STETKAR: I'm glad we choreographed this so well.

MR. HOLSTON: So any other questions on time of flight method?

Okay, next slide.

All right so there is two ways that the licensee would go into deeper analyses and possibly come to the point where they had to do time of flight and further reviews. They are going to conduct 50 destructive examinations. They are going to destructively examine 25 welds that have backing rings. They are going to destructively examine 25 welds that don't have backing rings. And they will judge that against a three-part acceptance criteria. What we are trying to verify with the Aging Management Program is that that root pass is less susceptible than the follow-on passes. And so the acceptance criteria is that there is no defect in the weld that will progress through 80 percent. You know so it's less than 80 percent of the depth of the weld. Dealloying is limited to 80 percent of the depth of the weld -- of the root pass. I'm sorry, the root pass. Thank you -- and if the phases that they discover within the root pass supports the basis and the basis is that there is no continuous beta or gamma-2 presence, in other words the beta or gamma-2 phases are surrounded by alpha phases, so if you can't get dealloying all the through the root pass.

If any one of those three criteria are not met, then they will proceed to do additional time of flight UT examinations to determine the extent of conditions.

And the time of flight UTs will be for each weld that fails destructive examinations. And then there will be periodic time of flight UT examinations of ten percent of the welds every five years. Now, when we are talking about the time of flights, as we discussed during the applicant's portion of the presentation -- this is of the above-ground welds -- so, ten percent. About 1600 of the welds are above ground, about 1800 of them are below ground. So they would be doing about 160 time of flight examinations every five years and, of course, accompanied by the immediate five more for every one that failed the destructive examination.

From those time of flight results, they can conduct a structural integrity analysis because the time of flight results will tell them the entire volume of the dealloying inside the volume. There is the same structural integrity analyses for the linear elastic analysis and the limit load analysis that we were talking about before.

I am going to pause for a moment on the time of flights and discuss another way that they could get into structural integrity analyses, as if they detect leakage in a weld.

The program will still require, just as they do today, for them to do a complete walkdown of all susceptible welds in the plant every six months and then, as you have read in the Safety Evaluation Report, they go out to the areas of the plant where there is buried piping and they look for water on the surface. As you read in the SER, that method has been accepted. We did extensive review of their calculations. And of course, actually, ASME Section XI allows you to conduct that type of examination for buried piping.

They will then do a destructive examination of that weld, if there is a weld that leaks, and then that plugs into the structural integrity analyses.

If the structural integrity analyses meet acceptance criteria, then that's as far as they go. We recognize that there is a leaker here. We recognize that over here may be a couple of destructive examinations didn't meet acceptance criteria. However, the structural integrity analyses are acceptable to the intended function will be met, which is our objective in License Renewal, reasonable assurance that the intended function will be met.

If the structural integrity analysis does not meet code, then we have a second decision to make and that is, is the weld operable. So we all recognize that the code is a boundary. We want to meet the code requirements because it gives you a whole lot of margin for meeting intended functions to systems.

However, just as we cover in Inspection Manual 326, you evaluate operability when you have a degraded condition. And if you can carry the structural loads with operability limits, then you're operable.

So if they pass weld operability, and we're at that final block at the bottom, then they are going to do a 95/95 sample time of flight UT examinations. And the 95/95 is around a couple hundred examinations.

And if the weld is not operable, then they are going to time of flight 100 percent of the welds -- of the above-ground welds. Now, how does that translate to buried welds? And that was asked during the applicant's presentation. It was essentially the same thing. They said there's a two-time margin on the allowable extent of dealloying with buried welds. So they will take the results of the volume from the time of flight and correlate that to here is all of our 1800 buried welds. Would that volume of loss of material affect a buried weld? And if it would affect a buried weld such that it would be not operable, then they will have to excavate and examine that weld with time of flight or cut it out and replace it.

We are reasonably confident that with the low seismic loads at the plant -- of course every day, there are dead weights being proven acceptable. And the further tolerance of the extended volume that could be loss of material in the below-ground welds, we will never come to that point -- in all likelihood, we will never come to that point.

The 95/95 sample, they will select welds based upon the construction details, potential consequences of failure. So they are going to basically risk inform to pick those that are of most risk to the plant. And that will be in the population of the 95/95 sample. Obviously, the 100 percent sample, they are just going to be looking at all the welds.

Timing of the inspections will be they

would get 20 percent of those time of flight examinations done within 30 days and the remainder within 180 days. And again, we are talking several hundred welds. The exact numbers are 246 examinations with backing rings and 262 without backing rings. That's what the 95/95 sample drives to.

MEMBER BALLINGER: I have two things. First, it is theoretically possible to operate the plant with a leaking weld for one entire cycle, basically. Is that correct?

In other words, if it meets operability --

MR. HOLSTON: Yes, if it meets operability, yes.

MEMBER BALLINGER: -- but it is a leaking weld --

MR. HOLSTON: Yes.

MEMBER BALLINGER: So that can happen?

MR. HOLSTON: Yes, sir.

MEMBER BALLINGER: The other concern is that prior to all of this happening, there has been an inspection done for your starting point, which presumes that you have made a selection of the welds that are the most susceptible to do the inspection.

MR. HOLSTON: Yes, sir.

MEMBER BALLINGER: If you end up with this situation, that means you have made a mistake because you now have a leaking weld which you didn't think was going to exist, based on your baseline inspection. So how do you do the 95/95 selection, when you know that there has been an error somewhere in the sense that you've missed a weld, which was previously inspected and found to be okay?

Maybe I'm not using the right terminology, but you get my point.

MR. HOLSTON: Right.

MEMBER BALLINGER: You've got a baseline which presumably is a sound baseline. If you get into this scenario, that probably means that your baseline was somewhere not sufficient.

MR. HOLSTON: Well, in any sampling-based program, we don't have absolute assurance that a defect -- all defects are going to be detected. And that, in essence, was the basis for the open item coming out of the subcommittee meeting. At the time of the subcommittee meeting, we didn't feel the corrective actions -- in other words, if you find a defective weld that doesn't meet acceptance criteria, we're robust enough. And that's where we drove to well, if you find one -- so now in your 50-sample, you have 50 destructive examinations, and you find --

well, even if you find one, okay, that one didn't meet the acceptance criteria. In other words, the root pass did not pass, did not pass acceptance criteria. That's why we have first, the structural integrity evaluation. And even if the structural integrity evaluation passes acceptance criteria and operability limits, we still are going to do the additional 95/95 sample, which, again, is a very significant sample. It's over 500 welds that are going to be time of flight examined. And the time of flight doesn't make any assumptions of what is going on. It interrogates the entire weld and seeks out where selective leaching is occurring.

So, if we find a leaking weld, is that a failure? Well, it's not a failure because sampling-based programs don't eliminate everything. You just have to have the adequate extent of condition reviews when you -- or extended condition inspections when you find that. That's why we use the 95/95. We're okay with that because if you are operable, operable means you can meet the intended function. And you can meet the intended function with a leaking weld, as long as it passes the structural integrity.

And we talked about that before, the linear elastic -- sorry about the linear elastic-plastic -- the linear elastic analyses, the limit load. So if you don't pass those operability limits, that's when we check 100 percent of the above-ground welds and then they will compare those results to all the below-ground welds. You know, again, they have more margin in the below-ground welds but they are still going to look at those with the worst-case extended dealloying that they saw.

MEMBER BALLINGER: So the backstop is the operability.

MR. HOLSTON: Yes, sir.

MEMBER RICCARDELLA: Bill, just a couple of questions just to help my understanding. The first box up there talks about these 25 percent destructive exams. If those don't find anything, there will be no nondestructive examination at all prior to the period of extended operation?

MR. HOLSTON: Well, there is, in addition to the destructive exams, it is 25 -- it's not 25 percent. It is 25 with backing rings and 25 without. That gives you about 90 percent confidence. That's what the numbers translate to.

There will also be -- I didn't talk about that here because we addressed it in the subcommittee, there will be 25 with backing rings and 25 without backing rings volumetric exams to check and confirm that they actually caught all the welds with flaws. Again, no absolute certainty. It is a standard-based program but it gives you reasonable assurance.

MEMBER RICCARDELLA: I understand. I just wanted to clarify my understanding. I was under the assumption it would be both destructive and non-destructive.

MR. HOLSTON: Yes, sir. Yes.

MEMBER RICCARDELLA: Okay. And then on the right-hand side of this chart, if you detect a leaking weld, you talked about what you would do, the sample expansion and all that. But I assume they will also repair that leaking weld at the next outage, right? I mean they're not going to let it operate indefinitely with a leak, right?

MR. HOLSTON: They will, yes, absolutely. And that's in the program, yes, that they will repair that weld. Now, if it's not operable, they are not going to be able to justify that. They'll still have to seek some code relief. Just because they have an Aging Management Program --

MEMBER RICCARDELLA: Yes, I understand but that's what they're doing now, right?

MR. HOLSTON: Right. Yes. Yes, that's

what they're doing now. So they would be replaced at the next outage.

MEMBER RICCARDELLA: Got it. Thank you.

MR. HOLSTON: Any other questions on the flow chart?

Okay, in closing, the staff reviewed the applicant's basis for the weld metal susceptibility and concluded that with the inspections that we discussed, the destructive examinations, the time of flights to determine the true extent of any issues of acceptance criteria aren't met and the corrective actions, there will be reasonable assurance that a loss of intended function will not occur as a result of loss of material due to selective leaching.

And with that, if you don't have any further questions, I will turn it over to Lois.

MS. JAMES: Okay, thank you, Bill.

Well, in conclusion, the staff has determined that the requirements of 10 CFR 54.29(a) have been met for the license renewal of South Texas Project Unit 1 and 2.

We will entertain any other questions you have.

MEMBER SKILLMAN: Colleagues, any questions?

Hearing none, Lois, Bill, Allen, Phyllis, thank you. Greg, thank you.

Dennis, back to you.

CHAIRMAN BLEY: Public comments?

MEMBER SKILLMAN: Are there any individuals in the room that would care to make a comment, please?

(No audible response.)

MEMBER SKILLMAN: Anybody on the bridge line care to make a comment, please?

(No audible response.)

MEMBER SKILLMAN: Back to you, sir. Thank you.

CHAIRMAN BLEY: Thank you. At this time, we'll be off the record until 10:45 but members, don't leave. I understand we have a letter for our consideration. We will start on that momentarily.

(Whereupon, the above-entitled matter went off the record at 9:31 a.m. and resumed at 10:45 a.m.)

CHAIRMAN BLEY: We are back in session for the 645th meeting of the ACRS. At this time, I am turning the meeting over to Professor Mike Corradini for our discussion of the NuScale Topical Report on the Safety Classification of Passive Nuclear Power Plant Electrical Systems.

VICE CHAIR CORRADINI: Okay. Thank you very much, Mr. Chairman.

So, the members may recall that we had a subcommittee meeting way back in March about this, a half-day meeting. And we discussed it over a range of topics, both in the open session, which we are now in, and closed session.

Before I turn it over to Omid, I would just remind everybody that we are going to go through this and staff will lead us, turning primarily to staff's discussion. We have NuScale available for questions, both physically and then, if need be, on the phone, but we are going to lead with the staff.

When Omid is finished with the open session, I will turn to look for any sort of public comments. And then will close it up, make sure everybody is bona fide to be in the room and open up a separate line to subject matter experts extra.

Omid.

MR. TABATABAI: Okay, great. Thank you so much, Dr. Corradini. Good morning, everyone.

As you mentioned, this is the full committee presentation on the subject of NuScale's electrical safety classification of passive nuclear power plant electrical systems or for short Electrical

Topical Report.

Back in March 2017, we briefed the subcommittee members and we had a half a day session. We got into a lot of good discussions and questions and answers. For the purpose of this meeting, I will go over -- provide an overview of the staff's review process and the conclusion. And the technical discussion will occur during the closed session, which will be probably five-ten minutes from now.

With me today I have -- Bob, if you don't mind introducing yourself.

MR. FITZPATRICK: Bob Fitzpatrick from the Electrical Branch, NRR.

MR. SCHMIDT: Jeff Schmidt from Reactor Systems.

MR. TABATABAI: Okay. In a nutshell, there hasn't been much change since we briefed subcommittee members. We pretty much are providing the same information but we have clarified a couple of conditions in the SER after briefing the ACRS' members as a result of your feedback and discussion with the NuScale staff.

Slide number 2, just a brief overview of the time line. We received the Topical Report back in October of 2015, the Revision 0 was submitted. And in

February 2017, we issued our SER for the Rev. 0 of the Topical Report.

NuScale submitted a Revision 1 to the Topical Report in February and we briefed the members in March. And in June of 2017, we issued our updated SER, based on Rev. 1 of the Topical Report. And our goal is to basically complete the SER by the end of August 2017.

Just a quick recognition for all of our staff who have been involved in the review of this Topical Review. As you can see, there have been 17 technical staff members reviewing this Topical Report and contributing to the SER in eight different technical areas.

The review process that the staff followed was basically, per request from NuScale, we reviewed the Topical Report for a generic passive design. We did not review it only for SMR design specifically. So, we wrote the SER. The staff wrote the SER for a generic plant, not for NuScale design.

We looked at the conditions of applicability in the Topical Report and we evaluated them against the applicable regulations and requirements of the NRC.

And also, the staff focused on the

reliability of the on-site DC power system and instrumentation for post-accident monitoring, reactor coolant system. And these are all based on the assumption that there are no Class 1E power available. So we just wanted to make sure that this function we had reasonable assurance that it would be achieved in the absence of a Class 1E power.

During the process, we identified six areas where we needed more information in order to complete our review. Four of those questions were related to the reliability of VRLA, the valve-regulated lead acid batteries and with respect to maintenance, design, and quality assurance provisions that are related to Reg Guide 1.155, which deals with station blackout.

Two questions were related to reactor safety, with respect to event non/escalation, and safe shutdown state.

NuScale provided responses to our RAIs and the staff found them acceptable.

And in Revision 1 to the Topical Report, NuScale updated the Topical Report and incorporated information that we had requested.

So in summary, the staff found that the Topical Report is acceptable to be referenced by an applicant, if they meet, basically, the conditions of applicability in those two tables that are in the Topical Report, 3-1 and 3-2. Plus, the staff has identified six additional conditions on the Topical Report and if an applicant can meet both of those conditions, these two items, conditions in the Topical Report and SER conditions, then they can reference this Topical Report in their license applications as part of their justification for not having Class 1E power system.

That concludes my presentation for the open part. If there are questions from the members or members of the public, this is a good time to ask.

VICE CHAIR CORRADINI: All right, I'll turn to the members. Walt.

MEMBER KIRCHNER: Let me see if I can frame this question so that we don't get into proprietary details.

In effect, aren't you, if you just look at your presentation, at the very high level, aren't we making a policy decision that you don't need Class 1E DC power for passive nuclear power plants, quote, unquote?

MR. TABATABAI: We discussed this question before and we and the staff decided, and they reached this conclusion that this is not a policy issue. It is a technical issue and can be addressed at staff level.

MEMBER KIRCHNER: I'm just looking at the precedent that you're setting. Assuming that other passive plants, let's leave NuScale out, are watching this proceeding and then saying well, if I have the attributes of a passive plant, then --

VICE CHAIR CORRADINI: As delineated in the table.

MEMBER KIRCHNER: -- as delineated, blah, blah, blah, then I don't need a Class 1 -- IEEE Class 1 DC power.

Now, in light -- I'm thinking of it from the perspective of say the public. Post-Fukushima suggests that it's a good idea to have reliable power so you know you're not completely in the dark.

Maybe my next statement was going to be the particular applicant goes to great lengths to show that they are proposing a comparable system in terms of reliability. But if I back away from the specifics, it's almost as if we're making a policy decision that quote, unquote, advanced passive nuclear power plants don't need that quality of DC power for all the reasons post-Fukushima that we are concerned about these kinds of issues.

So it's more an observation than a question.

MR. TABATABAI: I can offer two statements in response to that, without getting into the details of NuScale design. With respect to Fukushima event, of course, once an application is submitted for any licensing action, as part of that license review, we won't look at the Fukushima requirements and the regulations that we currently have with respect to that. So that is separate from this topic.

And with respect to future passive designs and applicants, when we get into closed session, there is one condition that the staff has already put on this SER that deals with that issue. We are not just basically giving a blank check here that hey, if you need this, then you are okay. But we go into the details of basically what --

MEMBER KIRCHNER: In closed session.

MR. TABATABAI: Right.

MEMBER KIRCHNER: But for the public, are they going to be aware of what that condition is?

MR. TABATABAI: Yes.

MEMBER KIRCHNER: And is that condition generic?

MR. TABATABAI: Yes. It is actually in the public version of the SER.

MEMBER KIRCHNER: Okay. All right, thank you.

VICE CHAIR CORRADINI: Other questions? Hearing none, let's turn to see if there is any public comment from the room. No public out there.

And people who are on the line, the line should be open. Is there any comments from members of the public on the bridge line?

MR. BROWN: It's open.

VICE CHAIR CORRADINI: Thank you.

Okay, hearing none, why don't we close the bridge line? And we're going to go into closed session. So that requires some high technology changes. So, everybody sit tight while our DFO has some fun.

(Whereupon, the above-entitled matter went off the record at 10:56 a.m. and resumed at 1:15 p.m.)

CHAIRMAN BLEY: The meeting is back in session, this 645th meeting of the Advisory Committee on Reactor Safeguards. At this time, I'm going to ask Dr. Ballinger to take us through the Advanced Power Reactor 1400 review work to date. Ron, please take the meeting.

MEMBER BALLINGER: Good afternoon, Mr. Chairman.

CHAIRMAN BLEY: Oh, and we're open session. I don't know if we got that marked. Go ahead.

MEMBER BALLINGER: Today we're going to hear from KHNP on Chapter 3, Design of Structures, Systems, and Components, etcetera, Chapter 4, the Reactor, Chapter 9, Auxiliary Systems, Chapter 15, Transient and Accident Analysis.

I think we probably have a meeting with two more chapters, and then we're finished with phase two. And I would like to ask if Bill or Mike would like to say something prior to our starting?

MR. WARD: I just want to say thank you for another meeting, and as you mentioned, we were hoping this would be the last, but there will be one more for phase three sessions. But we look forward to presenting the information, and to hopefully getting a good letter. Thank you.

MEMBER BALLINGER: So I'll turn it over to the folks.

MR. SISK: Well, this is Rob Sisk, Westinghouse, and again, I think you for the opportunity to meet and present the four chapters. Without any undue delay, I would like to introduce Dr. Hangon Kim as our presenter for the chapters to lead us through, so please?

DR. KIM: Good afternoon, my name is Hangon Kim from KHNP. I'm the project manager of the APR1400 design certification project. Today I would like to present a brief summary of DCD Chapter 3, Design of Structures, Systems, Components, and Equipment, Chapter 4, Reactor, Chapter 9, Aux Systems, and Chapter 15, Transient and Accident Analysis.

If there are questions or comments, I will try to answer the question. If I can't, our expert staff in this room will answer the question.

The first section of Chapter 3 provides a high-level description of how the plant structures, systems, and components important to safety meet the general design criteria in each individual subsection. In the next section, the classification of SSCs is described. The SSC classifications consist of a seismic category, quality groups, nuclear safety class, and codes and standards. All items are confirmed.

In this slide, I'd like to talk about the wind and tornado loading which is considered in the design of seismic category one and two structures. The design wind loadings on the surfaces of seismic category one and two SSCs subject to wind are determined in accordance with ASCE/SEI 7-05.

The APR1400 is designed to protect the SSCs against tornados and hurricanes. Maximum speed of design basis tornado and hurricanes are calculated according to the Reg Guide 1.76 and 1.221 respectively. For design basis tornado, the maximum tornado wind speed is 430 by power. For design basis hurricane, the maximum wind speed is 406 by power.

This section discusses the flood protection from external and internal flooding. The design basis flood level is designed in accordance with Reg Guide 1.59 and ANSI/ANS 2.8. The flood protection measures from external sources are designed in accordance with Reg Guide 1.102.

Internal flooding evaluation is performed for the reactor containment building and aux building by considering the water volume and flood over area. Structure enclosure, a barrier wall, drainage systems, emergency overflow line, and the watertight doors are designed to protect safety-related SSCs.

This section discusses the protection from internally or externally generated missiles. Safety related SSCs are protected from internally generated missile, turbine missile, missile generated by tornados and extreme winds, site proximity missile, and aircraft hazards.

MEMBER KIRCHNER: May I interrupt you for a minute for clarification? A few moments ago, you mentioned the tornado wind speed and the hurricane wind speed. Were those kilometers per hour, not miles per hour? I thought I heard 400. What were the numbers again?

DR. KIM: 230 and 260.

MEMBER KIRCHNER: Oh, 200. Thank you.

DR. KIM: Structures used to protect safety-related SSCs meet the requirements of related Reg Guides. Missile barriers are designed with sufficient strength and thickness to prevent local damage, including perforation, spalling and scabbing, and overall damage.

This section discusses the protection against the piping rupture to meet the GDCs two and four. High and moderate energy fluid systems are summarized in table 3.6-1. Separation, physical barrier, or pipe whip restraints are key to protect essential SSCs from the effect of postulated pipe break.

Postulated break locations such as terminal ends and intermediate break are determined in accordance with the BTP3-4, Part B. Non-conservatism of jet impingement model in ANSI 58.2 is addressed as an open item. The current technical report and the related REI will be revised to address these issues.

Break exclusion criteria is applied to ASME Class 2 piping in the main steam valve house between the containment wall and aux building anchor wall beyond the isolating valve. Dynamic and environmental effects due to the High and Medium Energy Line breaks are summarized in the pipe rupture analysis report.

This slide presents the APR1400 seismic design. The certified seismic design response spectra are defined as 0.3g, and the design time histories generates using requirements of option one, approach one in SRP 3.7.1. The generic site condition consists of eight soil profiles and one fixed-base condition.

For the seismic analysis model and method, 3-D finite elements models are developed and complex frequency response analysis method with ACS SASSI version 2.3 software is used for soil structure interaction analysis and the fixed-base analysis. Both the uncracked and cracked concrete stiffness cases are considered in the seismic analysis.

Structure-soil-structure interaction analysis is performed to evaluate the interaction effects between non-seismic category one structure such as containment building or compound building and the seismic category one structures. Incoherent SSI analysis is performed with hard rock high frequency seismic input motion which are set to 0.46g.

There are four Category 1 building structures in APR1400, Reactor Containment Building, Aux Building, EDG Building, and Diesel Fuel Oil Tank Building. The containment is a pre-stressed concrete structure composed of a right circular cylinder with a hemispherical dome and it's founded on a common basement.

A quarter-inch thickness liner plate is attached to the inside of the containment as a leak-tight membrane. The internal structures are physically independent of the containment except the supporting foundation basement.

There is one subcommittee question about the tendon temperature effects in the containment structure. The question is as follows, "Long tendons may be affected by ambient temperature change or temperature change during normal conditions and accident conditions inside containment. How is it considered in the structural analysis?"

Our response is as follows, "During structural analysis of the containment building, temperature change is not considered for the post tensioning system consisting of horizontal and vertical tendons since the effect of temperature change is negligible."

"The expected elongation of the longest tendon due to maximum temperature variation of the containment during accident conditions is 1.2 inches. It is quite small compared to the tendon elongation of 51.6 inches induced during stressing of the tendon."

"Furthermore, thermal expansion coefficient of tendon material is similar to that of concrete. The strains of concrete and the tendon due to temperature variation are also similar. It means that the tendon is extended due to the temperature variation is almost the same as concrete expansion. Therefore, temperature change is not considered since the effect of temperature change is negligible."

This section provides the method of design, dynamic testing, and analysis for Class 1, 2, and 3 components and supports including the classes , modifications and structures. The following information are provided.

Design transients used in the design and fatigue analysis, design loading combinations for components and component supports, dynamic testing and analysis due to pipe vibration, thermal expansion, and dynamic effects, especially a Comprehensive Vibration Assessment Program is conducted for the internals.

This slide provides the discussion on the equipment qualification of mechanical and electrical equipment. Equipment qualification is divided into a seismic qualification and the environmental qualification.

Seismic qualification confirms to GDC 3 and Reg Guide 1.100 and IEEE Standard 344. Environmental qualification is consistent with the rated requirements.

APR1400 equipment qualification program specifies the scope and requirements of the equipment qualification, qualification methods, documentation requirements, and the environmental condition of each room.

For the piping system design, a graded approach is applied. The scope of the design for ASME Code Class 1 piping includes RCS main loops, pressurized surge line, DVI line, and shut down cooling line.

And the scope of the design for ASME Code Class 2 and 3 piping includes main steam and main fuel piping from the nozzle of the steam generator to the main steam valve house penetration anchor.

Piping systems and supports are designed in accordance with the 2007 Edition with 2008 addenda of ASME Section III, Subsection NB, NC, and ND based on the 10 CFR 50.55a.

MEMBER SKILLMAN: Please back up one slide. For the RCS main loop, surge line, direct injection line, and your shutdown cooling, how are the incore pressure boundary lines addressed, incore lines?

DR. KIM: Do you mean the incore structure?

MEMBER SKILLMAN: No, I mean the lines that run from the bottom of the reactor vessel to your seal table.

DR. KIM: Do you mean the I&C cable?

MEMBER SKILLMAN: Yes.

DR. KIM: It's an instruments cable.

MEMBER SKILLMAN: Well, it's pressure.

DR. KIM: It's included in our graded approaching.

MEMBER SKILLMAN: Okay, it is part of the reactor coolant system pressure boundary?

DR. KIM: You're right, but in the design certification boundary, we selected this piping.

MEMBER SKILLMAN: The main components.

DR. KIM: Yeah, main components.

MEMBER SKILLMAN: Okay, thank you.

DR. KIM: The materials of threaded fasteners are selected to satisfy the requirements of ASME Section 3 NCA, NB, NC, and ND. These fasteners are designed and fabricated in accordance with ASME Section II and III, or Code cases allowed by Reg Guide 1.84.

Lubricants are selected to satisfy the requirements of NUREG-1339. Preservice Inspection and Inservice Inspection will be performed in accordance with the relevant requirements of ASME Section XI.

This is the end of Chapter 3. From this slide, I'd like to present DCD Chapter 4, Reactor. This slide shows the brief characteristics of APR1400 reactor core. Rated core power is 3,983 megawatts thermal. APR1400 has 241 fuel assemblies and each assembly has a 16 by 16 fuel rod lattice. The number of control element assemblies is 93, active fuel length is 12.5 feet. The maximum peaking factor is

2.43, and the maximum rod burnup is 60 GWD/MTU.

This slide explains the fuel assembly and fuel rod design. All design requirements and NRC guidelines are satisfied. As a point of the experience fuel, Pool Side Examinations and hot cell examination results showed that the PLUS7 fuels were well irradiated. 4,997 PLUS7 fuel assemblies have been supplied commercially as of 2016 since 2006.

MEMBER BALLINGER: I have a question. How many failures have you had? How many leakers?

MR. JEONG: This is Jae Hoon from KEPCO Nuclear Fuel. Among the 5000 fuel assemblies, we had ten assembly failure.

MEMBER BALLINGER: Ten leakers?

MR. JEONG: Yes.

MEMBER BALLINGER: Thank you.

DR. KIM: For the end of the life seismic analysis, the fuel assembly, grid, and the flow water damping tests were completed at EOL condition. EOL seismic analysis results show that the grid is not crushed. The EOL Seismic evaluation result will be submitted by the end of this month as a revision of the technical report.

This slide explains the nuclear design and the thermal-hydraulic design. The APR1400 is designed to satisfy the following requirements, power distributions are maintained within the design limits throughout normal operations.

Reactivity coefficients are maintained negative during power operation. Control systems are capable of providing enough shutdown margin and of controlling power distribution oscillations.

The DNBR limit was determined now to occur on a pure rod having the minimum DNBR at least of 95 percent probability with a 95 percent confidence level during stead-state operation conditions and AOOs.

1.29 of the DNBR limit was generated using the KCE-1 critical heat flux correlations that was approved by NRC coupled with the TORC subchannel code analysis, analysis code.

COLSS and RPS provide reasonable assurance that the design bases are not violated for any steady-state operating conditions and AOOs.

Okay, from this slide I'd like to explain DCD Chapter 9, the auxiliary systems. First, the fuel storage and handling section consists of the Critical Safety of New and Spent Fuel Storage, Spent Fuel for Cooling and Cleanup System, Light Load Handling System, and Overhead Heavy Load Handling System.

New fuel is stored in stainless steel

racks installed in a dry pit. Spent fuel is stored in a stainless steel rack with neutron absorbing material installed in a spent fuel pool filled with borated water.

The spent fuel pool cooling system is designed to maintain the spent fuel pool temperature below 60 degrees Celsius in a single active period. All piping penetrating the pool is located at approximately three meters above the top of the spent fuel assemblies, and all piping extending down into the pool has a siphon breaker hose above this level.

Light Load Handling System means the fuel handling system. Overhead Heavy Load Handling Systems consists of a containment puller crane and the pure handling area over the crane.

This slide is for the subsection of water systems. The major water systems consist of the Essential Service Water System, Component Cooling Water System, Ultimate Heat Sink, and the Chilled Water System. The Essential Service Water System transfers heat from the Component Cooling Water System to Ultimate Heat Sink. The CCWS moves heat from the safety-related components required for plant emergency shutdown and mitigation of design-basis accidents.

The Ultimate Heat Sink is a site specific system using the cooling tower since the cooling tower system is provided as a conceptual design for APR1400 design certificated staging.

The COL applicant is to provide Ultimate Heat Sink-related design information based on specific site characteristics including the conditions. The Ultimate Heat Sink provides cooling capacity for at least 30 days without breaker water.

The Chilled Water Systems consists of the essential Chilled Water System and the Plant Chilled Water System.

This slide is for the subsection of process auxiliaries. The Normal Parameter Sampling System takes RCS samples, shutdown cooling system samples, CVCS samples, and the primary off-gas samples. The Post-Accident Sampling System takes reactor coolant and containment atmosphere samples during post-accident conditions.

The CVCS is designed to perform the following functions, the reactor coolant chemistry and purity control, RCS inventory control, Boron recovery, RCS Boron concentration control to compensate the reactor changes, pressure control via pressurized spray, RCP seal injection, and the continuous removal of noble gases and other dissolved gases from the RCS.

This slide is for the subsection of heating, ventilation, and air conditioning systems. The major HVAC systems are the Control Room HVAC System, Engineered Safety Feature Ventilation System, and the Reactor Containment Building HVAC and Purge System.

The Control Room HVAC System provides adequate protection against airborne radioactivity and smoke from the outside, and this system limits the radiation exposure to the personnel in the control room under accident conditions to meet the GDC19.

The Engineered Safety Feature Ventilation System prevents possible accumulation of the oil fumes within the EDG area. This system maintains the hydrogen gas concentration to less than one volume percent in the battery rooms, and maintains the auxiliary building area under a slightly negative pressure with respect to the surrounding area.

The Reactor Containment Building HVAC System is designed to maintain the temperature of containment, ICI Chase, and the reactor cavity during normal operation and the loss of offsite power.

The Reactor Containment Building Purge System provides the property atmosphere and adequate ventilation for personnel before and during periods of personnel access, and this system controls the containment pressure.

MR. CINTRON: Mr. Kim, on the purge system, you have a high volume purge system that's normally operated during shutdown modes. You also have a low volume purge system, and the DCD said that it's operating during normal plant power operation when required.

And I think we asked during the subcommittee meeting based on your operating experience in Korea, what fraction of time is that low volume purge system in operation during power operation?

MR. SEO: This is Sung-Je Seo, KEPCO E&C. Let me explain the answer. The relevant reactor containment building low volume purge system is operated intermittently, not continuous.

MR. CINTRON: Yes, and because it's not continuous, I know the answer is not 100 percent of the time, and because it's intermittently, I know it's not zero, so it's somewhere between zero and 100 percent. I was asking based on your experience, where between zero and 100 percent that number might be. Is it ten percent of the time, or 70 percent of the time, or what percent of the time?

MR. SEO: The low volume purge system depends on the containment pressure and the radioactivity level.

MR. CINTRON: Yes.

MR. SEO: So without the personnel access inside containment, there is almost not operated low volume purge system. However, if an operator had to assess the inside containment, in that case, low volume will be operated before and during the personnel access. There is no exact time for low volume purge system.

MR. CINTRON: Well, I was just asking do you have low volume purge systems on other similar plants in Korea and what is the operating experience for some of those plants? I'm familiar with these types of systems, and it depends on the operating philosophy of the plant.

It depends on frequency of containment access for inspections and things like that, and some plants operate them continuously because they want to keep it cool and clean in there. Other plants operate them sporadically, and I was curious what your experience is.

DR. KIM: Okay, we will check our experience and then we will answer later.

MR. CINTRON: Thank you.

DR. KIM: Okay, this slide -

MR. CINTRON: By the way, the question, I understand how the system works. It's not a question about the system design. It's more related to some of my questions on the risk assessment and how the system is treated in the context of releases for the risk assessment. I don't really have any questions about the system design per se.

DR. KIM: This slide is for other aux systems. The major aux systems consist of the Fire Protection System, the Emergency Diesel Generator System, and the Gas Turbine Generator Facility. The Fire Protection System is designed in accordance with Reg Guide 1.189 and NFPA codes.

Separate redundant trains of safety-related equipment by three-hour fire-rated barriers for safe shutdown capabilities. Also this system maintains a 100 percent design capacity fire pump with one electric and one diesel-driven fire pump assuming failure of one fire pump or a loss of offsite power.

The Emergency Diesel Generator System is designed to provide for the required storage capacity and continuous supply of fuel oil to each of four redundant Class 1E EDGs following a loss of offsite power.

The Gas Turbine Generator Facility is designed to provide the standby power source for coping with station blackout in accordance with the requirements of 10 CFR 50.63 and Reg Guide 1.155.

Okay, from now I'd like to present Chapter 15, transient and accident analysis. Before we start the main body of Chapter 15, I will briefly summarize the Thermal Conductivity Degradation items.

APR1400 uses FATES3B code for the fuel performance analysis. However, FATES3B code cannot explicitly model TCD effects. Therefore, we decided to add fuel centerline temperature penalty into the FATES3B fuel temperature output based on a comparison with the Halden test results.

The technical report with the old information was withdrawn. The details about the TCD application are described in the revised PLUS7 topical report which will be submitted the end of this month.

MEMBER KIRCHNER: May I ask, since we haven't seen that, how did you take that penalty to compensate for the TCD? Did you just put it linear or did you try and match the Halden data?

MR. JEONG: Okay, this is Jae Hoon Jeong from KEPCO Nuclear Fuel. We had a lot of discussions with the steps and we compared our current results with the Halden test data and the other best estimate close measures, and we were able to get a certain amount of pure centerline temperature.

MEMBER KIRCHNER: Now, the Halden test results, don't they have - there's a thermal conductivity if I remember correctly. It kind of sags in the middle. It changes with burnout and temperature. How does that factor into your neutronic analysis? Is this on the margin or would that result in significant change to things like the thermal feedback with the reactor kinetics for the accidents that you analyzed?

MR. JEONG: This is Jae Hoon Jeong again. Actually TCD effects on nuclear design such as reactivity, there is no significant impact on that. For safety analysis, we have to consider TCD effects because peak cladding temperature includes LOCA.

We have to consider, you know, that similar effect on large break LOCA. So we have applied the penalty barrier on our code, and we have performed analysis using the EPRI for large break LOCA analysis and that penalty barrier, and all other safety results which are implicated by TCD, we redid analysis, and we're going to submit our results to the

NRC.

MEMBER KIRCHNER: Okay, thank you.

DR. KIM: The DCD sections impacted by TCD, which is simple text modification are 4.3 nuclear design, 9.1.1 critical safety of new and spent fuel storage, and 15.0 general information of safety analysis. And the full revisions are 6.2 containment system, 15.4.8 CEA ejection accidents, and 15.6.5 large break LOCA.

The analysis results for the full revision of DCD confirmed that all events and accidents concerned TCD satisfied the acceptance criteria. All DCD markups will be submitted by the end of this month, and the PLUS7 and the large break LOCA topical reports are also revised and markups will be submitted at the end of this month.

Let's start with the transient analysis. For the transient analysis, the main accident analysis is performed by CESEC-III codes and the calculation is performed by CETOP code. For the increasing heat removal by secondary system, the inadvertent opening of a steam generator relief of safety valve and the steam line break are quantitatively analyzed. The analysis results show that the minimum MDNBR remains above the fuel design limit, and the post-trip returnto-power doesn't occur for system line break.

For the decrease in heat removal by secondary system, the loss of condenser vacuum and the feedwater system pipe break are quantitatively analyzed, and the analysis results show that the RCS and the main steam system pressure increase, but below acceptance criteria.

For the event of reactor coolant pump rotor seizure and shaft break, COAST, HERMITE, CETOP, TORC, and CESEC-III codes were applied. Quantitative analysis for each event were performed, and it was confirmed that all the safety parameters meet the acceptance criteria.

The spectrum of CEA ejection accidents analysis applying TCD penalty was also performed, and the maximum system pressure, peak radial average fuel enthalpy, and the doses at the site boundary meet the acceptance criteria.

For the transient case of increasing reactor coolant inventory, the CVCS malfunction such as pressurized reactor coolant system malfunction are quantitatively analyzed. The analysis results show that the system pressure remains below the acceptance criteria.

For the transient cases of decreasing

reactor coolant inventory, the failure of small lines carrying primary coolant to outside containment and the steam generator tube rupture are quantitatively analyzed. Analysis results show that the minimum MDNBR remains above the fuel design limit, and the radiological acceptance criteria are satisfied.

Large break LOCA analysis, in APR1400, we've developed best estimate with certain quantification methodology, CAREM. The details of CAREM are described in large break LOCA topical report. In this model, RELAP5 and COMTEMPT4 codes are applied for system calculation and the minimum containment pressure is practical.

TCD penalties for the fuel centerline temperatures are considered. The final results with TCD are briefly shown below. PCT is 1,303 K, and peak localized oxidation is 6.3 percent, and the CWO satisfied the LOCA acceptance criteria. It is concluded that the results of LBLOCA with TCD satisfies the acceptance criteria.

For CENPD-137P conservative evaluation model with CEFLASH-4AS, COMPERC-II, STRIKIN-II, and PARCH is applied to the small break LOCA analysis. The licensing PCT is occurred in the DVI line break. The final PCT is 1,639 degrees Fahrenheit, and the results satisfy the acceptance criteria.

Post LOCA long-term cooling basically applies the CENPD-254-P conservative evaluation model. APR1400 is adopting the interim method which is applied to Waterford Unit 3. It is confirmed that the result of LTC satisfies the acceptance criteria.

This slide shows the design targets and design features for the dose analysis of design basis accidents. The does targets of the EAB and LPZ are taken from the 10 CFR 52.47, and according to the SPR, the limitation can be used for each DBA cases. The MCR habitability is ensued by applying the criteria in GDC 19.

The design features to minimize accident releases are as follows, safety injection system to prevent fuel damage, aux feedwater system for steam generator cooling, and so on.

The radiological analyses were performed based on the alternative system and the dose criteria of total effective dose equivalent. For the dose evaluation, RADTRAD code was used based on the conservative atmospheric dispersion factors. Analysis approaches are consistent with Reg Guide 1.183.

Doses to the public at EAB and LPZ for all DBAs are well within the dose limits. MCR habitability is ensured for all DBAs by complying the GDC 19. Okay, this is the end of my presentation. Thank you for listening to my presentation.

MEMBER BALLINGER: Thank you. We're a half-hour ahead of time. I hesitate to even say that because we'll probably fix that shortly, but are there any other questions from the members? Are the staff ready to go? So take a few minutes and switch out, and then we'll just pick up.

(Pause)

MEMBER BALLINGER: I'm told that there's a staff person on the phone line, and what we're going to do is to keep the phone on mute until a question comes up, if one does come up, and then we'll unmute it. Otherwise, we'll be listening to snap, crackle, and pop for the whole presentation.

MS. TERRY: Good afternoon. My name is Tomeka Terry, and I am the Chapter Project Manager for the APR1400 design certification application review for Chapter 3, Design Structure and Systems and Components and Equipment. Today, I will discuss an overview of the ACR Subcommittee on June the 5th, 2017.

In Section 3.71 and Sections 3.73 review, the evaluation Certified Seismic Design Response

Spectra. The development of the target PSD function should be priority generated from the Design Response Spectra other than Reg Guide 1.0 spectral shapes. The functions of the verification verified directions and development base on one-time scale of a horizontal target PSD function.

Acceleration time. Design time historic was found to be high-frequency consistent. Updated PSD function should be exceeded the minimal target of the PSD function.

In Section 3.72 review, the soil-structure interaction, SSI, sensitive study, the applicant evaluate the effects of separation of the soil from the sidewall since the two Poisson's Ratio and basemat uplift. The structure-soil-structure interaction analysis, SSI, the applicant performed an SSI analysis based on the embedding foundation configuration. The latter was pressure components for the SS, as well as the SS analysis, was higher than the dynamic soil pressure that was originally used in the design external below-grade wall in auxiliary building and in diesel or tank room.

The applicant re-evaluated the structure design in the external below-grade wall to consider the calculation, the maximum layer from the SSI analysis. The structure design Category I structure, the staff confirms the applicant methods for determining design adequate of the structure as consistent as NRC regulatory requirements.

Oxygen generator pressure load. The applicant determined that the structure integrity of the containment structure or the hydrogen pressure load meets the NRC regulatory requirements.

Ultimate Pressure Capacity. The applicant committed to using Reg Guide 1.216, Design and Acceptable Criteria to Determine the UPC of the Construction Containment.

Leak Chase Channel. The applicant committed to using the Leak Chase Channel system to monitor potential leak of the water from the RRWST.

Dynamic Lateral Earth Pressure. The applicant revised the structure analysis of auxiliary building and the DFOT to use a dynamic earth pressure obtained from the SSI analysis for governing the dynamic earth pressure.

In Section 3.85 and Section 3.74, the Tendon Gallery was included in the analysis design as part of NI component basemat. Waterproofing membrane would be used external below-grade horizontal and vertical surface of the structure of the APR1400 design.

The construction sequence in differential settlements. The applicant did not include a superstructure of the reactor containment building and auxiliary in the construction sequence evaluation and clearly do not determine the seismic type. The staff is currently addressing this issue.

Seismic instruction instrumentation. Seismic instruction instrumentation. In Section 3.92 and Section 3.95 review, the dynamic test analysis Comprehension Vibration Assessment Program reported that APR1400 steam-generated for induced vibration reactor designs compared to System 80 reactors design.

The reactor pressure vessel internals. Comparisons were made between the APR1400 design and the CE System 80+ reactor design. The reactors, as designed, are similar.

In Section 3.10 and Section 3.12 review, the seismic dynamic qualification equipment verified procedures evaluate to affect the hard rock high-frequency response spectra. The applicant will update this procurement specification to address the staff audit findings.

Pipe analysis and support to identify the environmental assessment fatigue for reactor coolant loop piping have not been performed. It has since been completed and currently under review.

The staff requests that the applicant information regarding the structure fatigue piping and pipe support could be impacted by vibrations or water hammering, which could potentially originate from operating the safe injection tank and fluidic device. The staff has issued an RAI to the applicant.

In Section 3.62 and Section 3.3 review, the determination of a rupture location and dynamic effects associated with the postulated rupture of the pipe, piping. The evaluation of blast wave and potential feedback amplification and resonance effects remains open. The staff is having a public meeting tomorrow with the applicant to discuss this issue.

Leak before break. The staff questioned the PICEP input file for the surge line fluid temperature against what is provided in the DCD. This issue remained open.

MEMBER STETKAR: Tomeka, Remind me. I think they're proposing to apply leak before break to the entire reactor coolant system, plus any connections to it; is that correct? It's not just the pressurizer surge line.

MS. TERRY: Let me get my staff. Eric

will answer that question for you.

MEMBER STETKAR: If you're going to say something, you have to come up to the microphone and identify yourself.

MR. SUNG: This is Ki Kwang Sung from KEPCO E&C. We apply it to everything in the surge line.

MEMBER STETKAR: Only the surge line or is it --

MR. SUNG: Surge line and shutdown cooling line, the lines more than the diameter, if the diameter is right and the 12 inches, we apply, except the main steam line.

MEMBER STETKAR: What I was trying to refresh my memory, and I think I found it, it's the leak before break is applied to reactor coolant loop piping, hot leg and cold leg surge line, direct vessel injection line, and shutdown cooling line. Is that accurate?

MR. REICHELT: This is Eric Reichelt from the staff. That is correct.

MEMBER STETKAR: That is correct. Okay, thank you.

MS. TERRY: And Section 3.91 and Section 3.93 review, special topics for mechanical components.

DPVIB computer program. The staff found the output of the DPVIB is in agreement with the test data. DPVIB computer program is added into the DCD Tier Two.

AC is in ASME Code Class One, Two, and Three components supports. Core support structure. Loading combination of the ASME Code Class One, Two, and Three. Components and component support conforms to ASME BPV Code, Section III.

In Section 3.96 review, the function design qualification and IST program. The staff conducted an audit to design specification APR1400 components in accordance with 10 CFR 52.47. The DCD description of the IST program based on ASME OM Code as incorporated in 10 CFR 50.55(a) and referred to the CO application. The staff would confirm by reviewing the DCD Revision 1.

In Section 3.11 review, the environmental qualification mechanical and electrical equipment. The staff reviewed the environmental qualification of the mechanical and electrical equipment to verify the equipment is capable of performing its design function under all normal environment conditions, the accident and post-accident environmental conditions.

Equipment qualification radiological. The accident doses are based on most limit design basis accidents for each area of the plant. Remaining outstanding issues including the doses within the AB during accidents do not appear to be adequate considering radiations throughout containment penetration. The staff requested the applicant provide additional information regarding how some of the post accidents gamma doses react rate information was determined.

This completes my presentation.

MEMBER BALLINGER: With regard to the last slide, well, slide number 11, I guess, that's request for additional information. Has that information been supplied? Is this closed?

MS. TERRY: No, it's not closed.

MEMBER BALLINGER: Not closed.

MS. TERRY: The staff has -- actually, the applicant has provided some additional information recent, last week, and the staff is reviewing that information.

MEMBER BALLINGER: So it's still an open item?

MS. TERRY: Yes, that is correct.

MEMBER BALLINGER: Thank you.

MR. WUNDER: Good afternoon, Mr. Vice Chairman and gentlemen of the Committee. I'm George

Wunder, and I'm the project manager assigned to Chapter 4 --

MEMBER BALLINGER: George, you need a green light.

MR. WUNDER: Thank you. I'll start again. Good afternoon, Mr. Vice Chairman and gentlemen of the Committee. I'm George Wunder, and I am the project manager assigned to Chapter 4, Reactor, for the APR1400 design certification review.

Staff's review of Chapter 4 addressed fuel system design, nuclear design, thermal and hydraulic design, materials, and reactivity control. We presented this chapter to the APR1400 subcommittee in February and, at that time, there were five open items. Two of these were associated with topical or technical reports in the area of fuel design. One was associated with our review of the instrumentation and control system and two related to materials. We had no staff actions for this chapter as a result of the subcommittee meeting.

Next slide, please. Our subcommittee presentation on fuel system design focused on the challenges we faced in the areas of burnup-dependent thermal conductivity degradation and fuel assembly structural design. In the applicant's original submittal, thermal conductivity degradation was not modeled, and this led the staff to be concerned about GDC 10 compliance. Burnup-dependent TCD is being addressed through a topical report. We expect the final revision of the topical report to be available to the staff for review by the end of the month.

We have an aggressive review schedule. Our due date for phase four input for this chapter is mid-September, but, if the revised topical is of good quality, we may be able to finish the review by the end of August.

MEMBER POWERS: Excuse me. Back to slide 12 of the five open items. Are they still open?

MR. WUNDER: Two of them are closed, those associated with materials. And I believe the rest are still open.

MEMBER POWERS: Okay. Thank you.

MR. WUNDER: Okay. In the area of structural response, the staff noted that the applicant did not rigorously adhere to the referenced methodology, and this called into question the load limit determination for the PLUS7 fuel. The applicant has developed a test program for the fuel for both the beginning of life and end of life conditions. The staff has conducted audits of the testing. Final technical and topical reports are expected by the end of the month. With an aggressive review, we may be able to complete the phase four evaluation by the end of next month.

Next slide, please.

MEMBER POWERS: Does aggressive mean sloppy?

MR. WUNDER: Never, never. Thorough always.

MEMBER POWERS: How do you do something more aggressively? Everybody works 16 hours a day or . . .

MR. WUNDER: They take very few breaks. Regarding nuclear design, the staff identified challenges in the area of control rod work depletion, the benchmarking of the nuclear design methodology, and the data associated with the nuclear design methodology. The staff conducted a series of audits of various calculations. The staff also conducted its own confirmatory criticality analysis.

As a result of its efforts, the staff was able to determine that all applicable design criteria had been met. The staff further concluded that the data and methodology employed are acceptable and are benchmarked appropriately. The staff's conclusions are supported by its own confirmatory analysis.

And the next slide, please. In our February presentation of the review of the thermal and hydraulic design, we talked about the many technical reports that we were reviewing in support of our Chapter 4 review. We noted challenges in the area of the core protection calculator system and the statistical combination of uncertainties.

Regarding CPCS, the APR1400 design is based on the approved System 80+ design, but the staff found the documentation in the System 80+ DCD did not provide enough information to resolve all of their questions. As a result, the staff conducted an audit and located the needed combustion engineering references. The staff determined that system functionality is based on changes that have been made and approved and implemented for the Palo Verde Nuclear Generating Station.

Regarding the statistical analyses, the staff determined that the methodology employed by the applicant does not conform to Regulatory Guide 1.105 Rev 3. This is an open item and is being resolved through the staff's review of Chapter 7, Instrumentation and Control Systems.

MEMBER SKILLMAN: George, should we

interpret the comment that you made relative to the Palo Verde site that the data for Palo Verde is identical to and will be used for the APR1400?

MR. WUNDER: I'd like to defer to, I think, Jim Gilmer for that one.

MR. GILMER: Good afternoon. Jim Gilmer, Reactor Systems Branch. Your question on the data for Palo Verde, the specific data will be developed during startup, as it was recently for the APR1400 design. I'm not sure if you, I grasped your question or . . .

MEMBER SKILLMAN: No, your response does not answer my question. George Wunder made the comment that, after locating the Palo Verde information, that this issue seems to have been resolved. And so my question is is the Palo Verde information identical to and will be used for the APR1400 design?

MR. GILMER: Okay. There were actual topical reports created for Palo Verde that are relevant to the APR1400, and some of these were also part of the CPC improvement that was also found in other plants. So these topical reports we are now making incorporated by reference in the APR1400 DCD. The actual data that will be used will be during startup testing.

MEMBER SKILLMAN: Fair enough. Okay, thank you. Thank you, George.

MR. WUNDER: Yes, sir. Let me have the next slide, please. Our February discussion of materials focused on material fabrication, controls on welding, non-destructive examination, the use of austenitic stainless steel, and degradation mechanisms. We identified open items in the areas and materials specification for the Versa Vent and in operating experience as justification for CRDM venting during refueling to keep oxygen levels low.

The applicant responded to the RAIs associated with these open items back in January. And since the subcommittee meeting, we have been able to determine that the responses satisfy the staff's concern. We now find that all applicable regulatory criteria have been met, and Section 4.5 may now be closed.

Next slide, please. With regard to reactivity control, the staff looked at control rod drive system functionality, environmental qualifications, CRDS cooling, single faults testing, and system performance. The staff did not identify any open items. The staff determined that all applicable general design criteria have been met and further determined that the CRDS testing requirements and technical specifications are appropriate.

This concludes our presentation on Chapter 4. If there are no questions, I will now turn it over to, oh, to myself for Chapter 9. And the next slide, please.

Good afternoon. I am still George Wunder, and I am the project manager assigned to Chapter 9, Auxiliary Systems.

The auxiliary systems can be divided into fuel storage and handling systems, water systems, process auxiliaries, heating, ventilating, air conditioning systems, and other auxiliary systems. I think the first draft of our safety evaluation contained about 25 or 30 open items. The safety evaluation we presented to support the May subcommittee meeting contained several open items. However, by the time we actually met with the subcommittee, I think that we had whittled it down to, I believe, five open items. There were no staff actions related to the, no staff actions on Chapter 9 as a result of the subcommittee meeting.

Next slide. Fuel storage and handled looked at criticality safety, spent fuel pool cooling and clean-up, and the handling of light and heavy loads. With respect to criticality safety, the staff found that the applicant used appropriate and accurate methodologies and models. The applicant analysis demonstrate compliance with the appropriate design criteria and regulations and that the staff's independent calculations confirm the applicant's analyses. An open item associated with TCD remains to be resolved, and, of course, we have to see the outcome of the fuel rack seismic analysis and make sure that no open items arise as a result of the staff review of Section 9.1.2.

With regard to spent fuel pool cooling and clean-up and load handling, the staff found that the system designs appear to be capable of meeting the stated design objectives. The staff is reviewing a bounding thermal analysis, and this should close the remaining open item.

Next slide, please.

MEMBER KIRCHNER: George, I missed in the SER the TCD thing that you mentioned. My intuition would tell me that TCD wouldn't be a big factor for the fuel pool. What was the issue there?

MR. WUNDER: Oh, I think that that goes to the fuel criticality analysis, doesn't it? I don't know if -- think this might be an Alex question.

MEMBER KIRCHNER: There is a margin in the heat transfer space. That shouldn't be an issue. I just missed why TCD would be a factor.

MR. WUNDER: We're going to have to unmute the phone.

MR. LU: Shanlai Lu from staff. Actually, what we're looking to is really the spectrum change, so, based on the information, based on whether that's a minor change --

MS. BURJA: Is it possible to unmute the line?

MEMBER BALLINGER: You're off mute now, Alex.

MS. BURJA: Okay, great. Sorry. I also wanted to --

MEMBER STETKAR: Before you start, identify yourself by name and organization.

MS. BURJA: This is Alex Burja from Reactor Systems.

MEMBER STETKAR: Thank you. Go on.

MS. BURJA: Okay. So to add to what Shanlai was saying, KHNP credits burnup in region two of the spent fuel racks, and one of the assumptions that goes into that is the maximum fuel temperature during operation because that will ultimately affect the reactivity of the fuel assemblies. So what we need to do for this review in terms of the TCD is to make sure that the effect of TCD didn't invalidate the maximum fuel temperature assumption for the criticality analysis. Does that answer your question?

MEMBER KIRCHNER: Yes. I would expect it to be small, but thank you.

MS. BURJA: Yes.

MR. WUNDER: Thank you, Alex.

MS. BURJA: You're welcome.

MR. WUNDER: And the next slide, please. Staff's review of water systems included the essential service water system, component cooling water system, domestic water and sanitary system, ultimate heat sink, condensate, storage facilities, and chill water system. The staff reviewed for compliance with the applicable design criterion regulations and concluded that the applicable GDCs are satisfied and regulations met. The staff concluded that there are no open items in this area.

Staff review of process auxiliaries included the compressed air and gas system, post-accident sampling system, equipment and floor drains, and chemical and volume control system. Staff reviewed for compliance with the relevant design criteria and regulations, as well as the appropriate regulatory guides. The review concluded that the design criteria satisfied and the regulations met, and the staff concluded that there are no open items in this section.

The staff's review of heating, ventilating, and air conditioning systems included the HVAC systems for the control room, spent fuel pool area, turbine area, engineered safeguard features areas, containment, and compound building. The staff reviewed for compliance with the applicable GDCs and regulations and found that the GDCs are satisfied and the regulations are met. The staff determined that there are no open items in this area.

Finally, the staff --

MEMBER BALLINGER: Back to the previous slide, 23. It says staff identified two open items.

MR. WUNDER: We're a slide ahead. That's the one I'm on now.

MEMBER BALLINGER: Oh, okay.

MR. WUNDER: The staff's review of Section 9.5 included the fire protection, communications, lighting, and EDG support systems, as well as the gas turbine generator system. The staff identified two open items associated with the communications system.

We have issued RAIs to address the open items. For the remaining sections, the staff has determined that the applicable design criteria and regulations have been met and there are no other open items.

This concludes our presentation on Chapter 9. I would now like to turn it over to Jim Steckel for his presentation on Chapter 15. Jim.

MR. STECKEL: This is Jim Steckel. Good afternoon, everyone. I'm here to discuss where we are with Chapter 15. Could I have the next slide, please?

Just as a small recap, these were our approaches to the review. We focused on the changes implemented into the APR1400 design from the CE System 80+ certified design and did an in-depth review of those safety issues identified after 1995 and provided overall coverage with the assistance of staff confirmatory analyses on selected areas. We identified potential issues early on and kept close communication with KHNP on our issue resolutions, and we continue to do that. And we used audits and on-site inspections to clarify certain issues.

Next slide. For Section 15.0, staff and KHNP have worked together on these two open items. One is boron dilution during LOCA. Long-term cooling responses have been submitted, and staff is in the process of updating the SER. And the second issue is the TCD. This has an impact on several FSAR sections. KHNP has just submitted the initial round of new calculations. Staff will meet with KHNP to review the details of the new calculations, and it is expected that the issue can be resolved according to the current SER schedule.

For 15.02, transient and accident analysis methods, the staff reviewed dozen of transient and accident computer codes. All were found acceptable except the pending large break LOCA topical report, and that review is scheduled to be completed this year and brought in front of the ACRS in December.

And for 15.03, radiological consequences, the calculated off-site dose results are acceptable, except for an open item regarding the control room and TCD dose results, and those are under review.

Next slide, please. For 15.1 and 15.2, these areas cover the increase or decrease in heat removal by the secondary system. The staff finds that the system response and analyses results are acceptable. 15.3, decrease in reactor coolant system flow rate. The system responses are considered acceptable and staff confirmatory analysis using TRACE/PARCS codes compare favorably in terms of major trends in overall system behavior. And Section 15.4.1-4, 15.4.1 through 15.4.4, reactivity and power distribution anomalies and startup of an active reactor coolant pump. The system responses are considered acceptable.

Next slide, please. 15.4.6, inadvertent decrease in boron concentration. The system response is acceptable, except for two open items. There's one open item questioning the conservative dilution times based on the complete mixing model for modes four and five with only one shutdown cooling pump in service. And the second open, in the second open we're questioning the CEA withdrawal event analysis assumptions.

15.4.7, inadvertent loading and operation of a fuel assembly in an improper position. The system response was considered acceptable. And 15.4.8, spectrum of Control Element Assembly Ejection Accidents. The system response is acceptable, except for the open item related to TCD.

Next slide, 15.5. In Sections 15.5, 15.6.1, and 15.6.2 and 15.6.3, these cover the transient of the increase of reactor coolant inventory, along with inadvertent opening of a pressurizer pressure relief valve, failure of a small line carrying primary coolant outside the containment, and steam generator tube rupture results. The staff finds the system responses analyses results acceptable.

And loss of coolant accident, 15.6.5, this section covers the loss of coolant accident and long-term cooling. The staff has a pending item for the large-break LOCA topical report review. For small break LOCA, all analyses are found acceptable except one open item regarding the justification of upper-bound break size selections. The staff has performed extensive review on the long-term core cooling analyses and testing. In particular, staff performed an on-site audit and inspection on KHNP's in-vessel downstream effects testing and their analyses. The final results submitted by KHNP are found acceptable, except for the open item on the loss of coolant accident deposition model, the DM model.

And, finally, the latest material submitted by KHNP on the LOCA DM model is being reviewed. The open items of loop seal clearing and boron dilution are being resolved.

And the final slide, please. For 15.7, radioactive material release from sub-system or component, the dose analyses are acceptable with respect to off-site consequences, but the response to the applicant's control room and TSC dose results remain under review at this time. And for 15.8, Anticipated Transient Without Scram, the evaluation is considered acceptable and our SER is being updated.

And this concludes our slide presentation. Are there questions?

MEMBER KIRCHNER: Jim, one of your colleagues, well, Clifford is working on a revised reg guide for reactivity-initiated accidents, and there are a number of plots in that guide that's out for public comment. I think maybe that is coming to an end. It's DG-1327. Where I'm going with this is that the plots that are there show burnup effects either measured in hydrogen uptake or other parameters that significantly lower the thresholds that are defined for failure below the 230 calories per gram that was cited in the applicant's analyses and just, one, are you aware of that; and, two, are you looking at the burnup effects that might impact that acceptance in terms of peak fuel . . .

MR. STECKEL: Someone much more familiar with it will answer. Shanlai Lu, please.

MR. LU: Shanlai Lu from Reactor System staff. We are aware that the reg guide will work.

That's the way you know that there is a burnup effect on that and that that's the reason, as part of a TCD actually, which is, you know, burnup dependent and then have a temperature history there, too. So, therefore, that's one of the items related to this section which related to radioactivity accident analysis, and that's an item in KHNP priority working on that and then we are supposed to get the results from that.

MEMBER KIRCHNER: That's what we'll see in July?

MR. LU: Yes.

MEMBER KIRCHNER: Thank you.

MEMBER BALLINGER: I have one final question, at least for me, and that is in these chapters, of all the open items, is there any anticipation that there will be an issue with respect to closing the open items for any of these chapters?

MR. STECKEL: Given enough time. No, we're determined to stick with this schedule without being sloppy, and we have paths forward on all the items, at least for 15.

MEMBER BALLINGER: But what about the other chapters? I'm asking that to everybody.

MR. WUNDER: We have paths forward and, if the responses that we receive are responsive to our questions and of reasonable quality, then, yes, we don't see any problems. I would like to raise one -- well, I don't want to call it an exception, but we have not yet presented to you a phase two for Section 9.1.2. So there has, we have a path forward on that, but we're going to present only a phase four product to you.

MEMBER BALLINGER: Okay.

MS. TERRY: For Chapter 3, we don't see an anticipation of anything going to hold us up from our schedule. We have a couple of RAI responses that we're getting back from the applicant as we speak now, so the staff is reviewing some things. I don't see any major road blockage at this point right now. I know we had that one issue with the jet impingement, which we have a public meeting tomorrow to discuss that information and, hopefully, it will give us what we need to move forward with our review.

MEMBER BALLINGER: Thank you. Questions from the Committee, others? While we're getting the bridgeline open, are there any questions from the audience? Hearing none -- oh.

MR. SISK: Yes, I just wanted to get back, we took an action to get back to a question during our session about the pressure purge that I just want to get on the record that we did take a look at the operating experience and, although plants varied across the board typically relative to the operation of low-pressure purge, looking at Member Stetkar's spectrum of never operating to 100 percent typically, typically, our values are in the neighborhood of about one to two hours a month to give you an idea of an operating cycle. During an operating cycle, averaging about one to two hours a month for the low-pressure purge.

MEMBER STETKAR: Thank you.

MEMBER BALLINGER: The bridgeline is open. Is there any comments from members of the public? Hearing none, thank you very much for your presentations, and we'll turn the meeting back over to Mr. Vice Chairman.

VICE CHAIR CORRADINI: We don't have a closed session you want to go into on the schedule?

MEMBER BALLINGER: No.

VICE CHAIR CORRADINI: Okay. So why don't we thank the staff and KHNP, and we'll take a break until ten of, and we'll come back and read a letter, somebody's letter. And we'll go off the record.

(Whereupon, the foregoing matter went off the record at 2:39 p.m. and went back on the record at 3:59 p.m.)

CHAIRMAN BLEY: We are back in session. At this time, I'm going to turn the meeting over to Jose March-Leuba for the discussion of the PAD5 work.

MEMBER MARCH-LEUBA: Right. Before we start, this is a closed session, so we want to make sure, number one, that the door is closed and that somebody make sure that there is nobody here that is not supposed to be. We're supposed to have one member from PNNL on the phone.

MR. GEELHOOD: Yes, I'm on the phone.

MEMBER MARCH-LEUBA: And you're on the closed line. Okay, perfect. So are we set?

(Whereupon, the foregoing matter went off the record and into closed session at 4:00 p.m.)

SOUTH TEXAS PROJECT

Advisory Committee on Reactor Safeguards License Renewal
Full-Committee Meeting July 12, 2017

Dave Rencurrel
Senior Vice President Operations

Agenda

- Introductions
- Station Ownership and Operation
- Site and Station Description
- License Renewal Application and Aging Management Programs
- Closure of Safety Evaluation Report Open Item
- Closing Remarks

Introduction

PERSONNEL IN ATTENDANCE

Dave Rencurrel	Senior Vice President Operations
Michael Murray	Manager Regulatory Affairs
Ron Gibbs	Manager Operations
Arden Aldridge	License Renewal Project Lead
Plant Staff	AMP Subject Matter Experts, Design Engineering Manager, Licensing, & Specialty Consultants

Station Ownership and Operation

Operated by STP Nuclear Operating Company (STPNOC)

STP Units 1 and 2 are owned by:

- NRG South Texas LP
- The City Public Service Board of San Antonio (CPS Energy)
- The City of Austin, Texas (COA)

Plant History & Major Investments

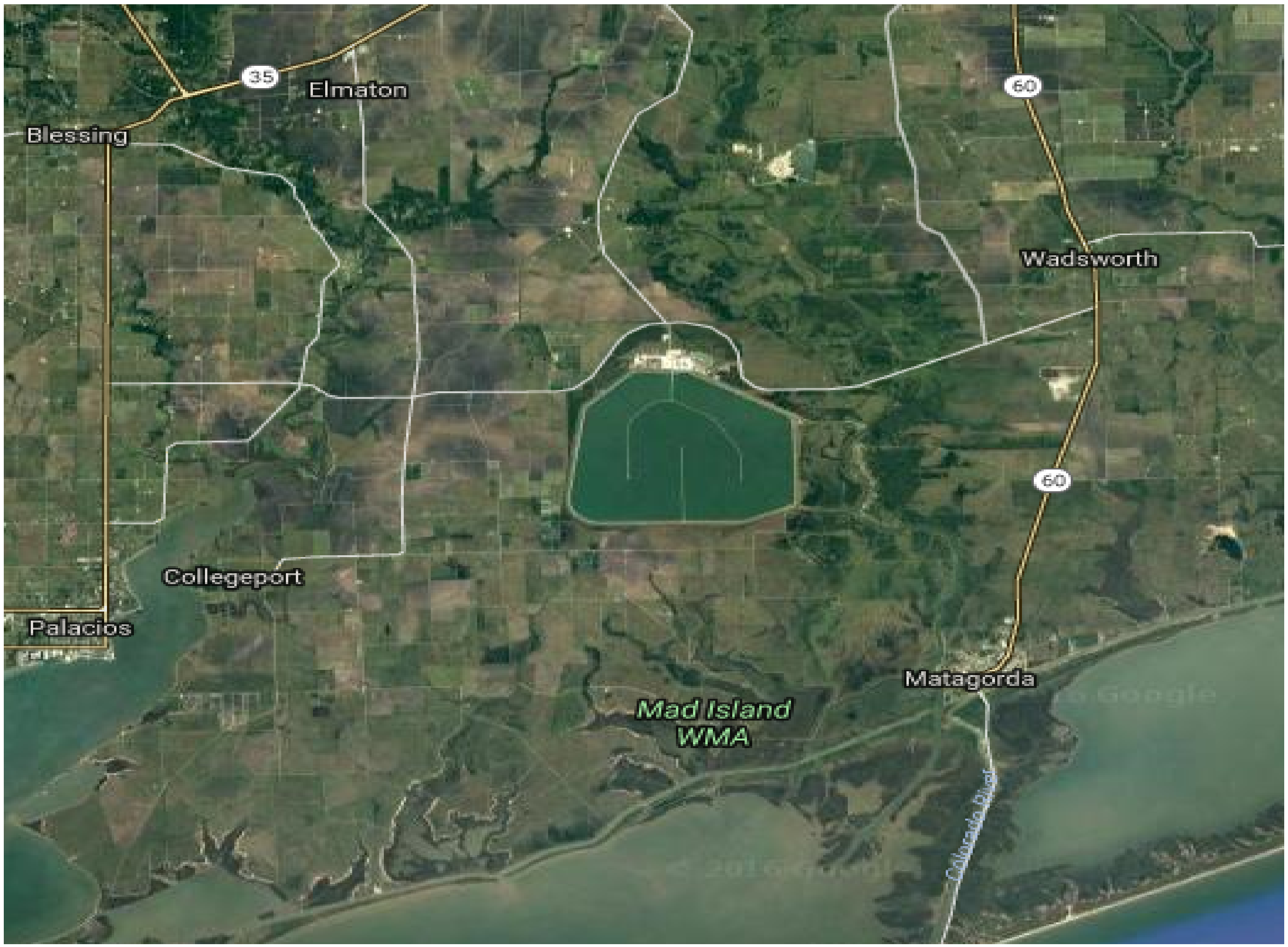
South Texas	Unit 1	Unit 2
Initial License	August 21, 1987	Dec 16, 1988
Steam Generator Replacement	2000	2002
Low Pressure Turbine upgrade	2006	2004
Replaced RX heads	2009	2010
Main Generator Stator rewind	2014	2012
Mechanical Stress Improvement Process(MSIP) (H/C leg nozzles)	2017	2019
Expiration of current License	August 20, 2027	Dec 15, 2028

Site and Station Description

Ron Gibbs

Manager Operations

SITE DESCRIPTION



STATION DESCRIPTION

License Renewal Application

Arden Aldridge

License Renewal Project Lead

License Renewal Application

License Renewal Application (LRA) submitted to NUREG 1801 rev 1	10/2010
NUREG 1800 and 1801 Revision 2 issued	12/2010
Annual Updates 2014, 2015, 2016, 2017	2014-2016
Issued safety evaluation report (SER) with open item	10/2016
ACRS Subcommittee meeting	11/2016
Issued final safety evaluation report (SER)	6/2017
ACRS Full Committee meeting	7/2017

GALL Consistency

Consistency Table

AMPS	AMPS Consistent	AMPS Consistent with Enhancements	AMPS Consistent with Exception & Enhancements	AMPS with Exceptions	Plant Specific
New (8)	3			4	1
Existing (33)	4	13	12	1	3
Total AMPS (41)					

License Renewal Commitments and Implementation

License Renewal commitments – 47 total

License Renewal commitments are included in UFSAR Supplement (Appendix A to the LRA) and managed through the STP Licensing Commitment Management and Administration processes.

Safety Evaluation Report

Since the ACRS Sub committee meeting:

- Clarified Selective Leaching of Aluminum Bronze Aging Management Program to close the SER open item:
  - Use Non-Destructive Examination to manage the age related degradation of aluminum bronze material and welds.

- Revised Steam Generator Aging Management Program to:
  - Incorporate LR-ISG-2016-01, “Changes to Aging Management Guidance for Various Steam Generator Components”

- Updated Bolting Integrity and External Surfaces Monitoring Aging Management Programs to:
  - Use inspection methods that detect leakage associated with closure bolting located in air-filled and gas-filled systems.

Closing Remarks

Dave Rencurrel
Senior Vice President Operations

Advisory Committee on Reactor Safeguards Full Committee Meeting

July 12, 2017

South Texas Project, Units 1 and 2
Final Safety Evaluation Report (SER)

Lois M. James, Senior Project Manager
Office of Nuclear Reactor Regulation

Presentation Outline

- **Overview of South Texas Project (STP) license renewal review**
- **Closure of Open Item (OI)**
  - OI 3.0.3.3.3-2: Insufficient details provided regarding applicant's Selective Leaching of Aluminum Bronze Aging Management Program
- **Conclusion**

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Review Milestones

- **License Renewal Application received October 28, 2010**
- **Safety Evaluation Reports (SERs) with Open Items issued February 15, 2013 and October 14, 2016**
- **ACRS License Renewal Subcommittee meeting held November 18, 2016**
- **Final SER issued June 8, 2017**

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SER Section 3

3.0.3 - Aging Management Programs (AMPs)

Applicant's Disposition of AMPs	Final Disposition of AMPs in Final SER
• 8 new programs – 3 consistent – 4 consistent with exceptions – 1 plant specific • 32 existing programs – 6 consistent – 13 consistent with enhancements – 3 consistent with exception – 8 consistent with enhancements and exceptions – 2 plant specific with enhancements • 1 existing program added - 1 plant specific	• 8 new programs – 3 consistent – 3 consistent with exceptions – 1 consistent with enhancements and exceptions – 1 plant specific • 33 existing programs - 4 consistent - 13 consistent with enhancements - 1 consistent with exceptions - 12 consistent with enhancements and exceptions - 3 plant specific

SER Section 3
Open Items Closed

OI 3.0.3.3.3-2: Insufficient details provided regarding applicant's Selective Leaching of Aluminum Bronze AMP.

- **<u>Concern</u>**:
  - The AMP did not adequately address corrective actions associated with inspection results demonstrating that structural integrity requirements would not be met.
- **<u>Resolution</u>**:
  - The program has been revised to address the open item.

Overview

Applicant developed a plant-specific aging management program to address selective leaching of aluminum bronze in its essential cooling water system.

Loss of material due to selective leaching of aluminum bronze can occur when:

- Aluminum content is greater than 8%
- The slow cool down rate of the casting or weld promotes formation of the susceptible β and γ-2 (lattice structure) phases
- The α phase is not susceptible

At STP:

- Susceptible castings ~ 350
- Susceptible welds (filler metal Al content) ~ 3400
- Piping material is not susceptible

Overview, cont.

- Since 1987 through 2015: 55 through-wall casting defects have occurred as a result of loss of material due to selective leaching.
- Since 1989 and progressing through 1994: 7 leaks in welds with backing rings. These leaks originated from weld defects that in most cases, progressed in part due to selective leaching.
- Based on testing by the applicant, the root pass of the weld is less susceptible to loss of material due to selective leaching because:
  - Faster cool down rate
  - Lower aluminum content

Applicant significantly revised the AMP in 2016:

- Susceptible cast components will be replaced.
- Susceptible welds joining nonsusceptible piping components will not be replaced.

TOFD UT Inspections

Time of flight diffraction ultrasonic method

- Detects circumferential extent and depth of dealloyed material (selective loss of aluminum from the lattice structure) within the inspection volume.
- The STP plant-specific procedure was developed in accordance with ASME Section V
- Staff review of the STP TOFD UT method
  - Validation tests
  - Implementation procedures
  - Personnel requirements

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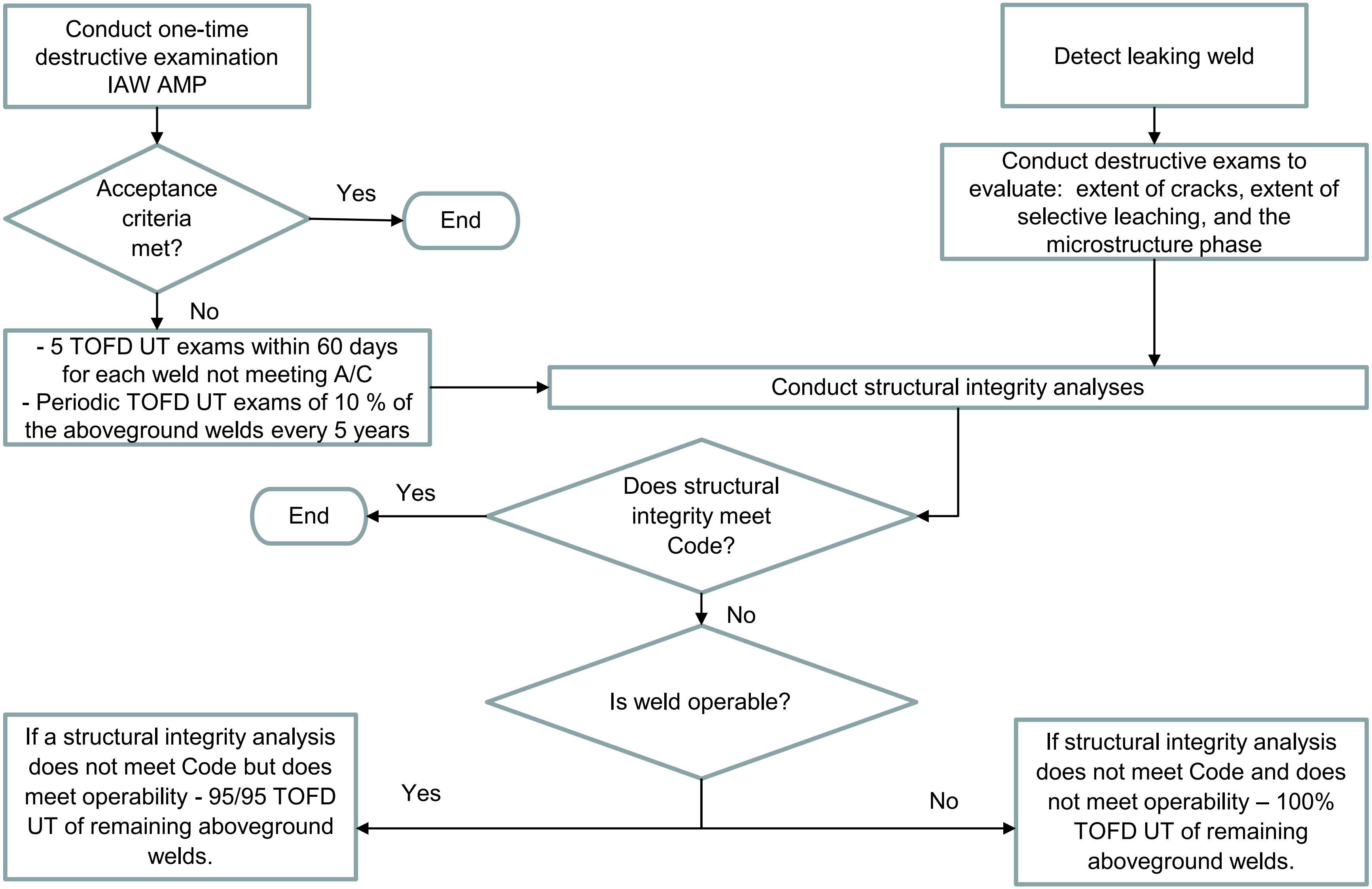
Corrective Actions



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Summary

The increased inspections provide the applicant with prompt insight into the extent of loss of material due to selective leaching:

- Number of other affected welds
- Extent of degradation within the welds.

This information will be used to assess Technical Specification operability of the system.

Conclusion

On the basis of its review, the staff determines that the requirements of 10 CFR 54.29(a) have been met for the license renewal of STP, Units 1 and 2.