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UNITED STATES NUCLEAR REGULATORY COMMISSION'S
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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644$^{TH}$ MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

OPEN SESSION

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WEDNESDAY

JUNE 7, 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 T2B3, 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

MARGARET CHU, Member

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JOSE MARCH-LEUBA, Member

DANA A. POWERS, Member
HAROLD B. RAY, Member
JOY REMPE, Member
GORDON R. SKILLMAN, Member
JOHN W. STETKAR, Member
MATTHEW W. SUNSERI, Member

DESIGNATED FEDERAL OFFICIAL:
CHRISTOPHER BROWN

ALSO PRESENT:
TONY AHN, KHNP
JIM ANDRACHEK, WEC/PWROG
DENNIS ANDRUROT, NRO
SURINDER ARORA, NRO
CLINTON ASHLEY, NRO
AYO AYEGBUSI, NRO
WILLIAM BAKER, TVA
DANIEL BARSS, NSIR
BEN BEASLEY, NRR
BRIAN BENNEY, NRR
STEVE BONO, TVA
MICHAEL BREACH, NRR
KATHRYN BROCK, NRR
MARK BLUMBERG, NRR

JOHN BUDZYNSKI, NRO
DENNY CAMPBELL, TVA
PAUL CLIFFORD, NRR
MATTHEW DEGONISH, WEC
JOE DEMARSHALL, NRO
ANTONIO DIAS, NRO
MICHAEL DICK, TVA
THINH DINH, NRO
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GERALD (GERRY) DOYLE, TVA
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TREY HATHAWAY, RES
MICHELLE HAYES, NRO

RAUL HERNANDEZ, NRO
VICTORIA HUCKABAY, NRR
DANIEL (LANG) HUGHES, TVA
SCOTT HUNNEWELL, TVA
SEOKHWAN HUR, KEPCO E&C
ATA ISTAR, NRO
DIANE JACKSON, NRO
HYEOK JEONG, KEPCO E&C
JOSHUA S. KAISER, NRR
SANGHO KANG, KEPCO E&C
JUNGHO KIM, KHNP
TAE HAN KIM, KEPCO E&C
YUNHO KIM, KHNP
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SHANDETH MONTGOMERY, ACRS
CHING NG, NRR
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ERIC OESTERLE, NRR
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JAMES ROSS, AECOM/KHNP
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ZUHAN XI, NRO

JINKYOO YOON, KEPCO E&C

SAMIR ZIADA, McMaster University*

*Present via telephone

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

P-R-O-C-E-E-D-I-N-G-S

8:30 a.m.

CHAIRMAN BLEY: The meeting will now come to order. This is the first day of the 644th meeting of the Advisory Committee on Reactor Safeguards.

Before we get to today's business, it's my pleasure to welcome Ms. Shandeth Montgomery to the ACRS staff as a program assistant. She joined the NRC back in 2008 and brings her extensive background in agency administration to the Committee. Her most recent assignment was in the Human Resources Office.

Welcome, Shandeth.

(Applause.)

CHAIRMAN BLEY: During today's meeting the Committee will consider the following: (1) Advanced Power Reactor 1400, the APR1400, the Browns Ferry extended power uprates, the Westinghouse proprietary topical report on PRA model for the Generation III Westinghouse shutdown seal, and preparation of ACRS reports.

The ACRS was established by statute and is governed by the Federal Advisory Committee Act. That means 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 the meeting is published. That said, we will also set aside 10 minutes for spur-of-the-moment comments from members of the public attending or listening to our meetings. Written comment are also welcome.

Mr. Christopher Brown is the designated federal official for the initial portion of this meeting.

Portions of the sessions on the APR1400, Browns Ferry extended power uprates and the Westinghouse proprietary topical report may be closed in order to discuss and protect information designated as proprietary.

The ACRS section of the U.S. NRC public web site provides our charter, bylaws, letter reports and full transcripts of all Full and Subcommittee meetings including all slide presentations at the meetings.

We have received no written comments or requests to make oral statements from members of the public regarding today's sessions.

There is a telephone bridge line. To

preclude interruption of the meeting the phone will be placed in a listen-in mode during presentations and committee discussions.

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

At this time we'll take up the issue of the APR1400 and I'll turn the meeting over to Professor Ballinger.

Ron?

MEMBER BALLINGER: Thank you, Mr. Chairman.

Today we will hear presentations by KHNP and the staff on Chapters, 6, 12, 13, 14, 16, 17 and 19. The schedule has evolved a little bit. Right now there will be no closed session. Everything will be open so that if it comes to a question or answer that involves proprietary information, the presenters will have to let us know so that we can make adjustments to the audience, if need be.

But now I'll turn it over to Bill Ward.

CHAIRMAN BLEY: Okay. And we'll hold those to the end of the session. If we need to go into closed session --

MEMBER BALLINGER: Okay.

CHAIRMAN BLEY: -- we'll do it at that time.

MEMBER BALLINGER: Okay. Fine.

Bill?

MR. WARD: Yes, thank you. The staff is happy to present to the Full Committee here. We're very anxious to keep moving. We have a 42-month schedule and everything's moving along pretty well. We think we have a short, but good presentation for today and we hope that we can answer all your questions. And thank you again.

MEMBER BALLINGER: Okay. So KHNP?

MR. Y. KIM: Yes, thank, Mr. Chairman and Mr. -- Dr. Ballinger for -- also for your review of all the ACRS members.

Today I am going to present for Chapter 6, 12, 13, 14, 16, 17 and 19. So let me start on Chapter 6 about engineer safety feature. So engineer safety feature materials are selected for compatibility with core cooling components and containment spray solution. The list of material is in the Table 6.1-1 and 2. The protective coating of APR1400 is listed in the Table 6.1-3 and they then meet Reg Guide 1.54.

For the containment system the containment of APR1400 is prestressed concrete with a cylindrical shell and dome. It is designed such that peak pressure is less than design pressure and well below half of the peak pressure within 24 hours after the accident. The leakage of containment is also below the design leakage rate.

For the APR1400 containment pressure and temperature analysis GOTHIC is used with assumption and initial conditions to maximize the containment peak pressure and the temperature. The analysis are performed for a total of LOCA and steam line break case feature selected for sensitivity study on barriers, break size and power levels.

For the containment sub-compartment analysis we analyzed at the place with use of COMPARE-MOD1A in response to high energy line break in each sub-compartment. Through the analysis we also performed the map.

The structure test is based on ASME CC-6000 and the leak-tight test is based on Reg Guide 1.163.

The containment heat removal system reduces pressure and the transient following steam line break and LOCA and the remove fission product from containment atmosphere following LOCA.

The function of combustible gas control in containment is to control the hydrogen concentration in containment and IRWST below 10 percent by volume during severe accident with PARs and igniters.

Let's move next, please. For the safety injection system this system consist of four independent trains and two diagonal trains. Trains 1 and 3 or trains 2 and 4 are required to mitigate the loss of coolant accident. APR1400 safety injection tank without fluidic device to eliminate the low pressure safe injection pump by extending the discharge time. KHNP got approval for the fluidic device topical report recently. The operability of safety injection system is checked through Tech Spec 3.5.1 through 5.

For the hability system the APR1400 are designed to allow control room operators to remain in the control room envelope and take action to operate the plant safely under normal and maintain it in a safe condition under design-basis accident conditions. And the APR1400 hability system protect control room operator from outside airborne radioactivity.

MEMBER POWERS: I am still unsure how you look at the dispersion under DBA conditions of radioactivity.

MR. Y. KIM: Could you -- I cannot hear you. Could you --

MEMBER POWERS: I am still unclear exactly how you calculate the dispersion of radioactivity under DBA conditions to the main control room.

MR. KANG: My name is Sangho Kang, KEPCO E&C In the DCD Section 2.3.4 we calculated the on-site dispersion of the LOCA condition of radioactive material, so based on the ARCO 96 model, which is developed by U.S. NRC, the concentration at the intake location of the main control room is calculated.

MR. Y. KIM: Yes. Now let me continue. So the exposure to MCR personnel meet the occupation dose limit of 50 millisievert. That is page 17 and Reg Guide 1.183.

Go to next, please. The APR1400 engineered safety features filter system are designed to mitigate the consequence of postulated accident by filtering radioactive particulate and iodine from air and meet the requirement of Reg Guide 1.52, and ASME N509, and ASME AG-1.

The filter system include control room emergency makeup air cleaning system, auxiliary building controlled air emergency exhaust system and the fuel handling area emergency exhaust system.

For the containment spray system TSPs stores at the holdup volume tank that enabled the IRWST remains above pH 7 during 30 days in LOCA condition.

MEMBER POWERS: I am not sure why I should be confident that the pH will remain above 7 for 30 days. Why are we confident of that?

MR. KANG: I think the -- our program's calculation to see the capability to maintain the pH of greater than seven was performed -- the purpose of this calculation to estimate the time to reach the pH greater than seven. But during the staff's review we got the comments that the -- can we make sure that the pH value can last for the whole duration of the accident? That's why we performed the analysis and the -- we performed that the pH value of greater than seven is maintained during the 30 days after the loss of coolant accident. It was responded by our RAI response to the staff's issue.

VICE CHAIR CORRADINI: Can we get the RAI number that we can just at least look at that?

MR. KANG: I don't remember now, but --

VICE CHAIR CORRADINI: That's fine.

MR. KANG: -- we can get it.

VICE CHAIR CORRADINI: If we can get it.

MEMBER POWERS: I would be interested in things like what you used for G values for nitric acid formation, G values for cable insulation decomposition, and especially how you treat phosphate precipitation by interaction with containments coming into the IRWST.

MR. KANG: So in order to calculate the pH value we have considered the adverse effect from the Hypalon which is the material of the cable. Also the HECR and nitric acid which comes from the composition material of the containment, inside the containment. We take into account all of this adverse impact materials.

MEMBER POWERS: Do you treat the conversion of organic contaminants to carboxylic acids like were seen in the RTF tests in Canada?

MR. KANG: I'm not familiar with that one. So we can back to you.

MEMBER POWERS: I would appreciate it.

MR. Y. KIM: Yes, I think RA number is found. So --

MR. OH: Yes, this is Andy Oh, KHNP Washington office. I think associated RAI number is 362-8445, question No. 31st.

MR. Y. KIM: Okay. Yes. In our LOCA

analysis the credit for aerosol and iodine -- elemental iodine removal by spray and for aerosol removal by natural deposition in containment business was credit.

Would you move next, please? For the in-containment water storage system that is consist of a storage tank and holdup volume tank and the cavity flooding system. The IRWST is protect, reliable and safety-related source of borated water for containment spray system and the shutdown cooling system and the safety injection system. The IRWST is also used to prepare the refueling pool in support of refueling operation.

Holdup volume tank collect and store coolant during the accident condition.

The function of a cavity flooding system is to flood the reactor cavity in the event of a severe accident.

And IRWST have minimum level of IRWST to support safety injection and containment spray pump operation during the accident condition.

MEMBER REMPE: Excuse me. When we discuss Chapter 19 we learn that even though you have this cavity flooding system, you did not take credit for it. You assumed the vessel failed to be conservative.

And it's still unclear to me how much you didn't take credit for it. In the PRA do you assume that the water still goes to the cavity instead of whether -- you know, when it's injected, even though you didn't take credit for the vessel to fail, did you assume good heat transfer from the vessel to the water? It wasn't clear how this evaluation was done.

And I believe when we discussed Chapter 19, you indicated, well, when we do severe accident management guidance, that will be something the COL applicant will take care of. So it -- that -- how the COL applicant will look at the certified design and take care of what actions should be considered is unclear to me. Could you give me any clarification on that interface?

MR. Y. KIM: Yes, let me check -- let me put it to our --

(Simultaneous speaking.)

MR. OH: This is Andy Oh, KHNP Washington office. I think we have a two feature of the mitigation feature. I think, Member Rempe, you mention that the thing is not that CF is -- cavity flooding system. That is IVREVC, in-vessel retention ex-vessel cooling system.

MEMBER REMPE: Yes, right.

MR. OH: That we didn't credit it, so it's a COL action item. COLAs, they will develop their strategy to implement that system, however, cavity flooding system is certified severe mitigation strategy. So it is included in our design.

MEMBER REMPE: So the PRA takes credit for that the water would be injected. And so what I'm kind of -- there's sometimes unintended concerns about external reactor vessel flooding. For example, if you don't take credit for it, do the staff still review to make sure that the structures are robust to withstand what would be needed for external reactor vessel cooling?

And so this interface puzzles me because if you say, well, we don't take credit for it, then you don't have to worry about is the insulation strong enough to withstand that chugging that would occur? When the APR1000 or the AP600 was looking at this, they had to consider how it operated.

And maybe this is also something the staff would clarify, but I'm just not sure how carefully the flooding system was reviewed and the strategy was reviewed, because you actually could have less heat transfer from the vessel.

MR. OH: Yes, all that is relevant to

support IVREVC, not the CFS system. But I --

MEMBER REMPE: What is the cavity flooding system for?

MR. OH: Cavity flooding system is just to open a valve from the holdup volume tank and provide water to the cavity. Probably it's a depth about five feet or six feet. But the difference from the IVREVC is IVREVC, we have to inject the water through the -- using a shutdown cooling to the cavity. And the flood level, probably it's more than 14 and 16, 17 feet. I should be covered. The bottom of the reactor vessel. So those two strategies totally different.

MEMBER REMPE: Okay. So I should have held my question until Chapter 19 then.

MR. OH: Okay.

MEMBER REMPE: But I will have that question when you get to Chapter 19 even though it's not in your slides how did you look at that? I understand now that it's a depth thing, but please clarify that when you get to Chapter 19.

MR. OH: Thank you.

MR. Y. KIM: This is end of the Chapter 6. In-containment water system we have a swing panel. Four vent stacks are located on the top of IRWST and three swing panels per vent stack are provided to protect overpressure and the vacuum of IRWST. Two swing panels per overpressure protection and one swing panel for vacuum protection. The right side -- the left -- to the right side figure shows the overpressure protection and the vacuum protection panel.

Those same panels are self-actuated by IRWST pressure and provided for hydrogen venting during the severe accident also. Those same panel also prevent vacuum in the IRWST during ECCS and the containment space inflation.

For the functionality of IRWST swing panels we got some request for additional information, so KHNP is trying to provide a response for this issue. So we are working on it.

VICE CHAIR CORRADINI: If I might ask -- I just want to make sure I understand the operation. So there's -- the one set of panels pull out inflow. Sorry. Thought it was on.

Let me just ask, I thought it was -- there is panels to the left of your mechanical drawing to allow inflow and then a different set to allow outflow. So does that mean one or both panels are always open?

MR. Y. KIM: Usually just they're slightly

-- slight close, but the mechanic can --

(Simultaneous speaking.)

VICE CHAIR CORRADINI: So what's the need of the panels at all if you're allowing both for in and outflow? I might be missing something about the function. Can you help me?

MR. Y. KIM: Yes, if you --

(Simultaneous speaking.)

VICE CHAIR CORRADINI: -- in and outflow by two different sets of panels.

MR. Y. KIM: Yes, you can --

MR. SISK: Rob Sisk, Westinghouse. Just to refresh on the panels, the primary reason for those panels of course is debris mitigation to the tanks itself and also concern for the condensation from the tank. So you want to keep the panels basically closed, but you don't want to have pressure differentials between containment and into the tank.

VICE CHAIR CORRADINI: So this is for water management then?

MR. SISK: Tank management, call it that if you will. You don't want debris and stuff going into your tank. You don't want condensation.

VICE CHAIR CORRADINI: Okay. I couldn't remember that.

MR. SISK: Yes.

VICE CHAIR CORRADINI: Okay. Thank you very much.

MR. Y. KIM: And then the next chapter is Chapter 12. I report it to Mr. Kang.

MR. KANG: Morning. My name is Sangho Kang, KEPCO E&C. I'm going to briefly talk about the overview of Chapter 12.

In order to ensure that the occupation radiation exposure ALARA APR1400 provides a structure to effectively implement radiation protection policy for any review that are consistent with the operational and maintenance requirement to Reg Guide 1.8, 1.33, 8.8 and 8.10.

The design policy of the APR1400 is to implement the ALARA velocity during the early stage of the design. The APR1400-specific ALARA design guide specify the approach, method and implementation guides for the designers to take into account. In the design of equipment the ALARA design guidelines applies to easy removal of contamination, enhancement of reliability to reduce maintenance and minimization of corrosion. And the layout design of the ALARA design guide requires to separate the radioactive equipment from the non-radioactive equipment and to provide sufficient areas for inspection and maintenance.

As for the radioactive source term, the reactor core fission products are estimated using ORIGEN-S computer code based on the thermal power of 102 percent. The source term of spent fuel is calculated based on 100-hour decay time before the movement from the core.

The reactor coolant system fission product source terms provided in Chapter 12 are determined based on the fuel defect rate of 0.25 percent using DAMSAM code and used for the design of shielding and ventilation. The CVCS source terms are calculated using SHIELD-APR computer program assuming that the gas stripper is not operated to maximize the gaseous sources.

The source terms of the shutdown cooling systems are calculating assuming that the systems stop operation at four hours after shutdown. The design base source system in the spent fuel pool water is determined assuming that the primary cooling water is mixed with spent fuel pool water after 48 hours of shutdown cooling operation.

The source term in the leakage waste management system are determined using DIJESTER computer code. The build-up activity in the gaseous radioactive system is calculated using the inflow from the CVCS tanks and the gas stripper. The solid waste such as spent-resins, filters, sludge are generated in the CVCS liquid waste management system, spent fuel pool cooling and clean-out system and steam generator blowdown systems.

The airborne activity in the radiologically-controlled area is calculated assuming maximum leakage of the radioactive components. The accident source terms which are used for post-accident shielding design and equipment qualification are determined by Regulatory Guide 1.183 which assumes a significant core melt.

The APR1400 design incorporates ALARA principles in accordance with Reg Guide 8.8 and 8.10. These principles apply to various areas of plant design. Design criteria for shielding as specified in Reg Guide 8.8, 40 CFR 190, GDC 19 and 10 CFR 50.34.

The shielding analysis are performed using several computer codes such as ANISN, MCNP and MICROSHIELD. The RUNT-G code is used to determine the post-accident shielding rulemakings. The shielding analysis produce design-basis drawings including radiation zone maps and the minimum required shield thicknesses.

The ventilation flows are provided to ensure the airborne contamination is maintained less than some of the DAC fractions. These air flows are designed to flow from the lower to higher contaminated areas so that the spread of contamination is minimized. In addition, the continuous air monitoring by effluent and area radiation monitors ensure the detection of the airborne contamination level change within 10 DAC-hours as is specified in the SRP 12.3.

Area radiation monitoring systems are provided to ensure to protect plant personnel from the unusual radiological events and monitor post-accident radiation levels. Some of the area radiation monitoring system provides balance of ESF, actuation signal.

In accordance with Reg Guide 8.19 the occupational radiation exposure for APR1400 was estimated based on the measurement data for operating 1,000 megawatt PWR in Korea. And this exposure data was increased taking into account the power ratio of APR1400. The resultant occupation exposure was estimated to be 585 person-millisievert per year or 58.5 man-ram per year.

The vital areas mission doses is estimated for both continuously occupied area and infrequent access areas. The exposure dose to the plant personnel who takes emergency action satisfied the regulatory limit of 15 millisieverts.

10 CFR 20.1406 requires to minimize contamination of facility and the environment and to minimize generation of radioactive waste. The detailed guidance in implement 10 CFR 20.1406 are provided in Reg Guide 4.21. To fulfill this requirement APR1400 design established several design objectives. Under these objectives the APR1400 design was reviewed and evaluated in accordance with the guidance in Reg Guide 4.21. As a result, an extensive list of design features are identified and provided in the DCD.

That's all for Chapter 12. If you don't have any more questions, I'll turn it over to Mr. Kim again.

MEMBER KIRCHNER: I have a question. So you went through all this. Based on your operating plants did you make any significant changes in material selection for corrosion or layout for maintenance or any other changes that might further your ALARA objectives?

MR. KANG: For this design of DC application the APR1400 design was based on the Shin

Kori Units 3 and 4. It's the first unit of APR1400. But the first Shin Kori design for the -- with respect to ALARA, the design for DC application is not so different from the Shin Kori example. But the Shin Kori 3 and 4 is quite different from the previous OPR1000 design with respect to the reduction of the corrosion and the layout of the plant. So all of these design features are provided in the Chapter 12.

MEMBER KIRCHNER: Okay. Thank you.

MEMBER BALLINGER: So to sort of clarify that in my mind, the exposure was based on the 1000 megawatt reactor, an OPR reactor?

MR. KANG: That's correct.

MEMBER BALLINGER: And those plants had different materials; in particular steam generator tubing material, those kinds of things, from the Shin Kori plants?

MR. KANG: Right.

MEMBER BALLINGER: And so why was there a correct -- why not use the Shin Kori plants as the basis for the ALARA as opposed to the OPR1000?

MR. KANG: Because the Shin Kori 3 and 4 didn't start operation yet.

MEMBER BALLINGER: Okay. All right. So that's that point.

MR. KANG: Okay.

MEMBER BALLINGER: Okay.

MR. KANG: All right.

MEMBER BALLINGER; So it didn't -- now, is there a plan to --

MR. KANG: Yes.

MEMBER BALLINGER: -- check it out?

MR. KANG: Sure. Sure. Once it's --

(Simultaneous speaking.)

MEMBER BALLINGER: To see if you're right.

MR. KANG: Yes, that's what we are waiting for.

MEMBER BALLINGER: Okay. Thank you.

MR. KANG: Okay. Thank you.

MEMBER SUNSERI: I have one other question. During our -- I get the chapters mixed up a little bit, but during our Subcommittee discussion we had a question regarding manual drawing of post-accident samples and whether or not those highly radioactive samples would be sufficiently shielded to keep the dose rates to the personnel acceptable. And it seems like that question was left hanging. So did you look into that and can you confirm that personnel are able to take the post-accident samples without undue radiation exposure?

MR. KANG: I think that question was not raised during the Chapter 12 Subcommittee meeting. I think some other chapter. But I think I can answer that question.

For the post-accident sampling we had special shielding design sampling. So we got the sample behind the shielding wall. So it's not the scope of the architect engineering design. That is provided by the vendor. And the sampling of the primary side or coolant is protected with respect to the radiation. So we take into account when we do -- when we calculate the vital area dose because the sampling area is one of the vital areas we need to take some -- the operator to take action. So when we calculate the dose to the operator for the sampling -- so we take that into account.

MEMBER BALLINGER: To expand on that again, have you considered related to Fukushima and their radiation doses in the post-accident sampling process and shielding here?

MR. KANG: Because we -- in our original design we already take into account the post-accident sampling during entry and exit for the sampling action, so we do calculate the dose to the operator during his passes and entry and coming back to the

MCR. So all of the information was provided in Chapter -- Section 12.4.

MEMBER BALLINGER: Yes.

MR. KANG: So, because it is required by the TMI action items.

MEMBER BALLINGER: Yes, but there's additional information from Fukushima.

MR. KANG: I think --

MEMBER BALLINGER: I think. Joy?

MEMBER REMPE: Ron, there were hydrogen explosions at Fukushima. They were BWRs. They were different power levels. I'm not sure that that would be relevant to this design.

MEMBER BALLINGER: But you have failed fuel?

MEMBER REMPE: Yes, they definitely had failed fuel, but again it's a lot different to going into those plants than what would be considered here.

MR. Y. KIM: I think because when we calculated the sampling dose, the post-accident sampling dose, we assume a significant core melt. And the IRWS is contaminated significantly. We take that into -- conservatively we calculate the dose.

MEMBER BALLINGER: Thank you.

MR. Y. KIM: Yes, let me take over Chapter

13. So conduct of operation. Chapter 13 consists of seven sections relating to conduct of operation. Most of Chapter 3 -- 13 are COL items. And the 13.6 is about physical security, which is SGI information, so we don't discuss today.

This chapter provide information related to preparation and the plan for the design, construction and operation of APR1400. Its purpose is to provide adequate assurance that COL applicant establish, maintain staffs of adequate size and technical competence. And this chapter shows that operating plant followed by license -- licensee adequate to protect public health and safety.

So let me just go the table. Proposed Section 13.1 is about organizational structure and applicant -- this is a COL item. And .2 is plant staff training. Also COL item. .3 about emergency plan. .4 is about operational program implementation. 13.5 is about administrative and operating procedure. 14.3 describe the physical security design piece of APR1400. The related information is in our security vault. So that is SGI information, so we cannot discuss that. Section 13.7 is about fitness-for-duty programs. Most of these are COL items.

Let me move to Chapter 14. Chapter 14 is about Verification Program. Chapter -- APR1400 Initial Test Program is developed to meet the Reg Guide 1.68. The test period is from the completion of construction to the power ascension test.

Chapter 14.3. ITAAC is not discussed today's meeting. That will be presented in the Phase 5.

The test phase divide into four different phase. Phase I is for pre-operation test. Phase II is about fuel loading and the pre-core hot function test. Phase III is criticality and the low-power physics test. Phase IV is a power ascension test. Total number of test we have 178.

Could you move next, please? For Chapter 16, that is about Tech Spec. Tech Spec of APR1400 are developed based on NUREG-1432. APR1400 is not applied risk informed tech spec. Unique design feature of APR1400 were reviewed for applicability of NUREG-1432 to APR1400. And related technical report was submitted to NRC.

The major deviation is APR1400 design adopt POSRV and the four safety injection trains rather than two trains and IRWST. And we have auxiliary feed water system with the turbine. And we have -- for the -- regarding electrical system, we have a four EDG rather than three EDG. So that is a major deviation, so that is incorporated in our tech spec.

The tech spec number is specified in my presentation.

For Chapter 17, Quality Assurance Program. Chapter 17 is composed of six section, Section 1, 2, 3 and 5 are about Quality Assurance Program. Section 4 is related to the Reliability Assurance Program, RAP. Section 6 is handling about the maintenance rule. KHNP is prepared to submit and submitted a QA Program to NRC as the form of a topical report. This comprise with all the regulatory requirements such as Part 50, Appendix X, Part 21 and ASME NQA-1. This QA Program is implemented and improved according to corrective action in KHNP.

Move next slide. As an applicant KHNP has the final responsibility to design control. KHNP delegated the design activity to three design suppliers: KEPCO E&C for balance of plant and NSSS design; KEPCO NF for fuel and core design; DOOSAN for NSSS component design.

To oversee the delegate design activity KHNP are doing five kind of major activity. First, KHNP evaluate, select and qualify supplier and seal the contract with them only. Second, KHNP receive, review, approve the design product from design supplier and the NRC. Third, KHNP control RAI and all the related change in the design process. Fourth, KHNP preside committee related to design change, interface and the project measurement meeting. Fifth, KHNP does QA inspection internally and externally to check the implementation status and to improve QA Program as a part of corrective action.

Let's move to RAP requirements. This slide is about RAP. The design, construction and the operation are consistent with risk insight and they key assumption from PRA. The programmatic control ensure that RAP list is appropriately developed and maintained and communicated to the related organization. QA Program oversee activity of affecting RAP SSC quality.

The RAP is implemented during operation phase via Maintenance Rule Program, QA program and the various test and maintenance programs. Design RAP should therefore align for a seamless transition to these programs.

So maintenance rule --

MEMBER SKILLMAN: So back up on, please. To your last bullet, there should be seamless transition, that transition will be seamless only if the leadership that is involved causes it to be seamless. Those of us who have been through this know that this is an effort that takes an enormous focus by many people. And so I would ask even though this plan is many years in the future, what KHNP's vision is of how this seamless transition will be ensured.

MR. Y. KIM: Yes, usually in Korea we have very many operate -- conceptualization project during the last more than 20 years. So in our project is three organization. KEPCO E&C, DOOSAN, and KEPCO NF and the KHNP work together every project for reports and the Kori unit that we have ability accumulate the experience to work together. So we have -- each company has their own energy expert. So through the many meeting with expert panel -- so we are going to update the RAP SSC. So probably we can do the seamless transition. So from the design -- I'm not sure that help you.

MEMBER SKILLMAN: That's in the right direction. Thank you.

MR. Y. KIM: So Maintenance Rule Program is also described in SRP 17.6 that is required by Part 50.65. The detailed procedure and the implementation will be the responsibility of the COL applicant. We are going to seamless transition to this program.

We have to move to Chapter 19, PRA. Chapter 19.1 covers the APR1400 PRA. Shown on this slide are the scope of PRA by the operation mode and the hazard. Note that seismic evaluation was performed using PRA-based seismic margin assessment and we applied a bounding approach for the level 2 internal flooding at low power and the shutdown mode.

This slide highlight the key approach for PRA modeling and quantification. The PRA use a linked fault tree method to ensure that initiate and support system impact on the front line mitigation system are considered. Common-cause failure was considered for the intra-system SSC. Additionally, inter-system CCF modeled between containment spray and the shutdown cooling pump because the pumps are identical.

Human error probability initially set artificially high to ensure dependent operator accidents are not prematurely truncated prior to application of a dependent HEP barriers. Quantification is performed using truncation value 10 to minus 13 for the CDF; 10 to minus 14 for the LRF quantifications.

Uncertainty analysis contains three part: parametric uncertainty; identification of a modeling uncertainty; quantification -- and the quantification of a selected uncertainty case. Parametric uncertainty use the state of correction to estimate the mean barrier of a CDF from point estimate produced by PRA quantification. Modeling assumptions were identified and evaluated for the potential impact on the conclusion. For sensitivity case two for CCF, two for HEP were quantified.

Let's move to next, please. Key design improvement from risk insight include original two EDG design change to four EDG. AAC, we change from diesel generator to gas turbine generator. Also we increased capacity for 125-volt DC battery. Also, Tech Spec 3.6.7 is revised for the equipment hatch closure in Mode 5. Also, the protection of cable from fire damage is the same.

Let's move next, please. Other application in DC Phase include also design RAP, SAMDA and the input to human factor engineering and the physical security. Operational Phase would include operational RAP, ROP, CVTR determination process and the maintenance rule.

The next slide shows that the overall risk profile from PRA. The result show that no one hazard or operating mode dominate the risk. Internal flooding risk is very low due to the design feature of the plant.

Next slide.

MEMBER STETKAR: Let me stop you there. It's my understanding that we had several questions about the PRA during the Subcommittee meeting and we were informed at that time that KHNP is updating the PRA to include several what I'd call substantive changes. I know that you're proposing to revise the models for reactor coolant pump seal failures. I think we were told that you're reviewing all of the operator actions in both the full power and shutdown models to ensure consistency in the treatment because then they'd be treated differently in different parts of the models. And there were several other items of varying potential importance to the overall risk and the distribution that you show on this slide.

Is that still true? Are you planning a substantive update to the PRA?

MR. Y. KIM: Yes, we are working -- still working on it. And could you help me with that?

MR. ROZGA: This is Greg Rozga from Enercon and the simple answer to your question is yes, we are working on that. We are at this point --

MEMBER STETKAR: Greg, get real close to that mic.

MR. ROZGA: We are at this point complete with the Level 1, and the Level 2 changes are going in now.

MEMBER STETKAR: Okay.

MR. ROZGA: And we -- for Level 1 we still have to do the dependency analysis, but we want to make sure that we get the dependency analysis between Level 1 and level 2, so we're kind of waiting --

MEMBER STETKAR: When you say "dependency analysis," you mean --

MR. ROZGA: HEP dependency.

MEMBER STETKAR: HEP dependency?

MR. ROZGA: Correct.

CHAIRMAN BLEY: Okay.

MR. ROZGA: Correct. So those actions in Level 2 we want to make sure if there's any dependency with the Level 1 actions that we capture that.

MEMBER STETKAR: You also -- when you say Level 1, you mean both full power and shutdown --

(Simultaneous speaking.)

MR. ROZGA: Correct.

MEMBER STETKAR: Because that was one of the bigger areas that you were concerned about.

MR. ROZGA: Yes. And then the --

MEMBER BALLINGER: And now for the complex answer. When?

MEMBER STETKAR: Well, I'm assuming that will require an update to the DCD.

MR. Y. KIM: Yes, we are going to provide that soon, so we will try very hard.

MEMBER STETKAR: Try very hard.

MR. Y. KIM: Okay.

MEMBER STETKAR: Well, we'll try very hard to review it in a timely manner then.

VICE CHAIR CORRADINI: So can I ask a question on that, John? Or, Ron, I should say. Excuse me.

So once this is done for Chapter 19, are we going to come back and have a Subcommittee meeting or wait until the DCD is presented to staff and staff has to -- what I'm trying to understand is the process at this point. I understand the process.

MEMBER BALLINGER: The current answer is no, that the next time we would see this would be in Phase V.

VICE CHAIR CORRADINI: Which is 18?

MEMBER BALLINGER: Yes. At which time people just take their chances that it'll work. That's the current -- and Bill can correct me if I'm wrong, but I think that's the current answer.

MR. WARD: That's the current answer as I understand it. I've had discussions with KHNP and at this point they didn't want to go forward with a second Subcommittee meeting.

MR. Y. KIM: Yes, I think that there is most organized, and so you are going to close all the organized probably.

MEMBER REMPE: So during our discussion we learned that the PRA said that bypass failures dominate core damage frequency. And when we started delving into that, we learned that you based your conditional probability for consequential steam generator tube rupture on work that was done for a Westinghouse plant. And there has been a more recent report done by the staff where they evaluated a CE geometry and a Westinghouse geometry and they found an order of magnitude higher for that consequential steam generator tube rupture. So that's an example of something that I think might be a significant change.

And during our discussion I believe KHNP responded that wasn't available when we did the PRA. And this is a new document. It was -- the draft is publicly available and they would look into it. Have you guys looked into it and said, well, maybe we should use a geometry that's more appropriate for CE design?

MR. DREMEL: Ray Dremel from Enercon. We have not been able to incorporate that at this -- as of yet into the model. We talked earlier. That document came out after we had started down Revision 0. We need to look into whether we can incorporate that and how that will be incorporated into the next --

(Simultaneous speaking.)

MEMBER REMPE: That might be a good example of something that might make a significant difference. And it would be nice to see that result. That you are planning to do that, is what I'm hearing.

MR. DREMEL: We will look into incorporating it. I'm not going to make commitments for KHNP on what they will or will not do, but that is something we need to consider.

MEMBER REMPE: Okay. Thank you.

MEMBER KIRCHNER: I'd like to ask a different line of questioning. I've always felt that the most value of the PRA is to insight it gives, not the numbers. So if I just look at your graph, it looks like half -- and I'll use the term loosely, half of your risk almost is due to fire and over half of the risk is at low power. So what are you doing --

MEMBER POWERS: With all plants.

MEMBER KIRCHNER: I know. I know. So my question is you have this information. Now what do you do with it? Are you content with this or do you then look at your operating procedures or your fire protection programs and look at where the vulnerabilities are here? How do you -- what are you using this for that improves your design?

MR. DREMEL: This is Ray Dremel with Enercon again. And roughly -- as you say, roughly half the risk is due to fire, but the total risk for APR1400 is fairly low.

MEMBER KIRCHNER: Right.

MR. DREMEL: So this is a big piece of a small pie.

MEMBER KIRCHNER: I just --

MR. DREMEL: And we have looked at -- as Mr. Kim mentioned earlier, there are certain cables that if we -- if they are protected and they're put in as a whole item, that greatly reduces risk.

We did use the results to -- what are the vulnerable -- what are the big contributors to risk? We eliminated -- we did make COLA commitments to reduce that risk. Now we're down to the -- the risk is fairly well balanced across things. There's not one --

MEMBER KIRCHNER: Dominant?

MR. DREMEL: There's not a silver bullet. There's -- as this chart shows, the risk is fairly well distributed across all modes and all events. Flooding risk is low because there's just no flood sources in the aux building.

MEMBER STETKAR: Just for clarification for the members here, that's internal flooding.

One thing I wanted to mention on the record is that this profile that we see here does not include risk from seismic events which are not quantified, nor does it include the risk from the so-called other external events categories. For example, high winds, tornadoes, external flooding and so forth. If those were included in the PRA, they might somewhat alter this profile. We don't -- there's no way of knowing that until the site-specific PRA is done with their hazard and winds and flooding and so forth. So this is the part of the picture that they can show at this point for the certified design.

MEMBER KIRCHNER: Thank you.

CHAIRMAN BLEY: I didn't attend the Subcommittee meetings, but one thing I've seen in other design certifications is for the kinds of locale-specific events that Mr. Stetkar just described. They often used a surrogate seismic source and surrogate water designs and then said at the COL stage you have to ensure you're less vulnerable than that surrogate or you have to do more work at that stage. Is that the same thing done here or did you just not consider those kinds of events at this point?

And if that second one is true, what direction is there for the COL when you actually know what the site-specific issues would be?

MR. DREMEL: There were screening analyses done for certain external events like high winds. It showed that we expect those based on conservative criteria to be a negligible contributor to overall risk. For seismic events we can't -- you can't really do a seismic PRA because you don't have a plant to look at. You don't know what specific make and model of components were done.

So if there is a COL item, the licensee would have to do a full seismic PRA before fuel load. Certain external events like external flooding are left to the COL applicant to show that they are within the constraints we've assumed in their screening analyses.

CHAIRMAN BLEY: That seems an appropriate time to look at those and I hope should there ever be one, we actually do that.

MEMBER STETKAR: There is just -- again for the other Committee members, like as you're aware, I took a pretty good look at what they've done. They do have a -- I would call it a more robust PRA model-based seismic margin analysis than we've seen in other design certifications. They do not have quantified seismic risk values, either core damage frequency or release frequencies. They have done a -- I'd call it a limited conditional quantitative evaluation where they've ranked seismic contributors by running a nominal seismic event through their PRA models, which is a little bit more than we've seen in other studies.

CHAIRMAN BLEY: That helps, but given what was just said, when the staff comes up, I'd be interested in their expectations for the COLA PRA.

MEMBER BALLINGER: This overall risk profile includes FLEX, or whatever you're calling it?

MR. DREMEL: No, we took no credit in this PRA for FLEX.

MEMBER STETKAR: Be careful. You took credit for some FLEX-y stuff, so you took credit for the --

(Laughter.)

MEMBER STETKAR: -- whatever you call the -- I've always forgotten the acronym, the truck that sprays the containment.

MR. DREMEL: Yes. Yes, the --

(Simultaneous speaking.)

MEMBER STETKAR: And that's kind of a FLEX-Y sort of thing.

MR. DREMEL: Yeah. So any credit would be minimal. As Mr. Stetkar said, it would be the external -- ESCBS.

MEMBER STETKAR: Yes.

MR. DREMEL: External containment spray --

PARTICIPANT: ECSBS.

MR. DREMEL: ES -- yeah. So we hook up a fire truck to an extra header in the containment to reduce containment pressure. But as far as the Fukushima modifications we -- the PRA took no credit --

(Simultaneous speaking.)

MEMBER STETKAR: Yes, the other parts that -- we'll hear about some of those I think in 19.3.

MEMBER BALLINGER: I'm cautioned by Mr. Stetkar about comparing to other plants because everything's different, but I have the risk poster for another CE, the System 80 plant.

MEMBER STETKAR: It's irrelevant.

MEMBER BALLINGER: That's what he cautions me on.

(Laughter.)

MEMBER BALLINGER: But I'll say it anyway. The fire risk is the same. The total fire contribution for that plant is pretty much the same as this.

MEMBER STETKAR: The fractional contributions here though at a high level are not unusual for a broad spectrum of plant designs, whether they're PWRs or BWRs or the country of origin or anything. This is not an unusual risk profile. And typically it will change somewhat once the seismic risk is quantified.

MR. Y. KIM: Then let me move to next chapter. Chapter 19.2 is severe accident evaluation. 17 -- 19.2 describe severe accident evaluation. APR1400 is designed to meet the related criteria such as Part 50.34 and Part 52.47, and the SECY-93-087, and the Reg Guide 1.216.

Comprehensive analysis of in and ex-vessel severe accident progression are made with conservative approach to cope the uncertainty. First, for the hydrogen there's -- hydrogen concentration inside the containment is lower than 10 percent by large free volume of containment and PAR and igniters. So the hydrogen combustion risk can be eliminated, can be minimized.

For the molten core concrete interaction this mid-line can maintains its leak-tightness by using cavity flooding system.

Could you move to next slide? For high pressure melt ejection and the direct containment heating reactor coolant system pressure at the reactor vessel failure can be low by rapid depressurization system and the possibility of containment failure from direct containment heating can be minimal.

For the fuel-coolant interaction reactor vessel and the cavity structure integrity is confirmed.

For the containment performance Reg Guide 1.216 Position 2 related with the combustible gas control and the Position 3 for more likely severe accident challenge are confirmed.

Accordingly, the analysis result indicate containment and the severe accident mitigation features of APR1400 meet the severe accident requirements.

MEMBER REMPE: So, excuse me. I think

this is the time to re-ask my question about the in-vessel retention system and the fact that it exists, but no credit was taken for it and the fact that we were told that that will be left to the COLA applicant.

My question is really at a higher level is -- this interface about how if an applicant came in, how they would extract what's need for the severe accident management strategies. Earlier in your slides today you said we are preparing the severe accident management strategies. Do they consider the in-vessel retention system? And is it in the PRA even though you don't take credit for it? Did the staff review the robustness of that system and what the effects are with water addition in their accident management strategies?

MR. OH: Yes, this is Andy Oh, KHNP, Washington office. And that matter is -- actually in-vessel is not credited as a mitigating feature for the APR1400 due to the uncertainty surrounding associated phenomena. For example, steam explosion or something. And along with the operator's action, for example for the shutdown cooling, have to operate the cavity flooding system.

So the position for the KHNP is for the

SAM, severe accident mitigate -- management system is when develop the COL applicant, they have to consider both the negative impact and the positive impact together. If they think it's for -- positive impact is more than negative impact, they can implement in their SAM, those severe accident management system. However, for APR1400 is -- this feature is already implemented in our design. We are not obligated for the COL applicant have to use this features their -- the SAM, severe accident management. So that is our position.

So probably in a COL applicant they can evaluate more of the steam explosion or other associated phenomena for the negative impact to that in-vessel, especially the strategy. If they think is more -- positive impact is more than the negative, they can implement that strategy to their management system.

MEMBER REMPE: Thank you. So now what I'm hearing is that it's an optional system at the -- and I'm wondering now; and maybe I should ask the staff, but did they review an optional system for robustness and possible adverse effects? And if they -- again, when this comes up with the applicant, it used to be clearly documented at this optional system wasn't considered in the review and if they want to implement, then they need to have it reviewed by the staff because it could have some negative impacts. And I guess I didn't get that from our earlier discussion or my review of the Chapter 19 --

(Simultaneous speaking.)

MR. OH: And also it's clearly described in the DCD and also in the PRA as in-vessel. Ex-vessel is not included in -- for PRA or de rigueur. It is also -- we just did the sensitivity study when it -- it is taken into account for the PRA how much the risk gain can be achieved or not. We just show them to -- but we didn't take into account for the system for credit, give the credit to our PRA model.

MEMBER REMPE: Okay.

MEMBER STETKAR: By the way, just for the record, for the other Committee members, this discussion has perhaps given people the implication that this is some separate system. It's a pipe with two valves in it that provides a water flow path from one train of the combined safety injection and shutdown cooling system return line into the reactor coolant system. Or you can align that line also to the boric acid makeup pumps, I believe. So it's not a separate stand alone system. It's a pipe with two valves in it. It's a flow path.

MEMBER REMPE: So --

MR. WAGAGE: This is Hanry Wagage from the staff. We have not reviewed the system because the applicant did not take credit for the system. In the PRA applicant did not take credit, but when we were reviewing steam explosion ex-vessel, we asked the question what would happen if the system failed and much larger melt pours in -- much larger diameter melt pours in the cavity and cavity would have much higher water level than steam explosion loading will be higher.

Then applicant proposed COLA item. That is -- for the COL applicant is to evaluate the effectiveness of ex-vessel cooling system. We have not reviewed this system so far because applicant need not take credit. However, there is an open item. We have an open item under that we are discussing with the applicant. It's not close yet.

MEMBER REMPE: So this fuzzy line of where the -- of what will be done will be clearly documented and what the staff finally produces to say we've not considered the adverse effects. If they want -- if an applicant does come in and says I'm going to try and invoke this, that brings in a bunch of analysis about steam explosions and possible adverse effects, the robustness of the insulation that would -- or the system that would have to allow water to come in, etcetera, near the vessel.

MR. WAGAGE: Right now that's the position. We have not reviewed. And that's what applicant is proposing, a COL information item. But that's still under review.

MEMBER REMPE: Okay.

MR. PHAN: Good morning. My name is Hanh Phan, the senior lead, PRA and severe accident reviewers. Yesterday the reviewers and the senior management, we had a meetings to discuss and come up with the possible resolution to address this issue. We decide that we will ask the applicant to send us their analysis, any assessment they performed to certify that the operation of the system would not impact the containment or any consequences to the plant. We will have another discussion with the applicant and we would ask them to send us their assessments. And the staff will reviews that.

MEMBER REMPE: Okay. Thank you.

MR. Y. KIM: So we are 25. So we are moving to 19.3 for the beyond-design-basis external event. APR1400 FLEX strategy was developed -- ELPA concurrent with loss of ultimate heat sink in accordance with EA 12-049 and 12-051 and NEI 12-06.

In order to implement the FLEX strategy, APR1400 design has various FLEX equipment such as three secondary pumps for cooling, two high-head pumps for reactor coolant system makeup, one pump for spent fuel pool makeup and another one pump for spent fuel pool spray. Also we have two FLEX pump for emergency core secondary bypass system, ECSBS, and two 480-volt portable GTG to restore one of the Class 1 E bus.

On-site resource include large amount of cooling water and a fuel oil source that will be used for 14 days and 30 days, respectively.

For the off-site resource we assume 4.16 kilovolt portable AC power and indefinite amount of cooling water and the fuel oil resource available.

MEMBER STETKAR: Mr. Kim, before we leave this slide, there's -- I want a couple clarifications. The second sub-bullet under FLEX says two primary low-head pumps. That's different from Rev 0 of the DCD, isn't it? Rev 0 of the DCD only said one primary low-head pump. I believe that's correct --

MR. Y. KIM: Yes, and let me just --

(Simultaneous speaking.)

MEMBER STETKAR: -- from my notes. I'm

just trying to find out where there's a change from what we saw previously in Rev. 0 of the DCD.

MR. OH: Yes, and this is Andy Oh, KHNP Washington office. And NEI 12-06 required n plus one requirement. That's for the -- in actual the strategy we used one, however, we prepare n plus one so that's -- two is the correct number.

MEMBER STETKAR: Two is the correct number? And also that's -- the same is true for the ECSBS pumps?

MR. OH: Yes, and per the previous ECSBS we prepared one.

MEMBER STETKAR: Yes.

MR. OH: And during a conversation with NRC after ACRS we corrected it to that.

MEMBER STETKAR: Yes, I just wanted -- okay. So I wanted clarification, because in my notes from Rev 1 of -- Section 19.3.1 only listed one primary low-head pump and one ECSBS pump. So that you are now committing to two of each of those?

MR. OH: Yes.

MEMBER STETKAR: Thank you. And this by the way is why I said ECSBS is FLEX-y-type stuff, because it is the -- of these pumps, the equipment on this slide, the PRA does include credit for ECSBS;

and, Greg, you'll have to correct me if I misspeak, and only ECSBS from this inventory. Is that right?

MR. OH: Yes, this is Andy Oh, KHNP Washington office. Originally ECSBS is not prepare for the FLEX equipment. That's originated based on a SECY-93-0178.

MEMBER STETKAR: Andy, I don't want to split hairs in terms of the name of the stuff. The pumps that are called ECSBS on this slide that has a FLEX heading on it are included in the PRA, is that correct?

MR. OH: Yes, correct.

MEMBER STETKAR: Okay.

MR. Y. KIM: Let's move next slide. Core cooling strategies at power operation is divide into three phases. In Phase 1 reactor coolant system hot standby condition is maintained with no operator action. Phase 2, reactor coolant system is a cool down to around the shutdown cooling system entry condition using installed or proxy. In Phase 3 reactor coolant system control state is maintained with the same strategies in Phase 2 and but using off-site resource.

For low mode operation with a steam generator available the same power operation strategy is applied after waiting until reactor coolant system heat up to hot standby condition.

For low mode operation with steam generate not available during Phase 1 decay heat is removed by reactor coolant system inventory boil-off. And in Phase 2 feed and bleed operation is applied using low-head FLEX pump. In Phase 3 the operator can resume shutdown cooling system operation or continue the same strategies as in Phase 2 using off-site water and fuel oil source. The detailed procedure will be developed by COL applicant.

MEMBER STETKAR: Mr. Kim, on this slide my notes say you've listed -- for at-power the Phase 1 coping time is eight hours. For shutdown modes my notes indicate that the Phase 1 coping time is three hours. Is that correct?

MR. OH: This is Andy Oh, KHNP Washington office. For the beyond-design-basis external even strategy we -- initially we developed the low-power shutdown strategy, however during the RAI process the staff request to the applicant for the portion of low-power shutdown strategy that should be into the COL action totally. So --

MEMBER STETKAR: What? Okay. We'll talk to the staff about that. Just for the record, Andy, are you saying that the staff asked you to remove that from the DCD?

MR. OH: No, and for -- removed this -- not the -- ask removed the DCD. That's NEI 12-06 in Revision 3 based on that for document and for low-power shutdown strategy can be deeper to the COL applicant. There's several requirement based on that revision because the --

(Simultaneous speaking.)

MEMBER STETKAR: We'll try to get clarification from the staff on this. I'm really confused because quite honestly; this is my personal opinion, I was quite pleased to show -- to see in the DCD that you had addressed strategies for shutdown conditions explicitly, which most people have ignored. So I was quite pleased.

And in fact the strategies show that there are different time windows and different things to do during shutdown. That's why I brought up this three-hour versus eight-hour. It's the three-hour only for the condition when you're at reduced inventory and the steam generators are not available. That's a minor -- it's an important but it's a detailed nuance.

And now I'm hearing that that may be removed because of some interaction with the staff.

So I'll be real curious to see how the staff addresses this. Thank you.

MR. Y. KIM: Yes, for the maintain containment function containment analysis results show that containment is maintained well below design pressure and EQ temperature limit for 16 days. In addition, ECSBS is available to manage the containment condition after reaching design pressure.

For the spent fuel pool cooling decay heat is removed by spent fuel pool inventory boil-off during Phase 1, but in Phase 2 and 3 the operator needs to make up spent fuel pool inventory using FLEX pump and the water it was from raw water -- raw water tank. And two channels level instrument -- with the two-channel level instrument the operator can monitor the spent fuel pool water level in MCR during the event.

Let's move to next slide. Next slide about loss of large area. The APR1400 design was buried in accordance with SRP 19.4 and NEI 06-12. Those strategies in NEI 06-12 are considered to maintain the spent fuel pool and the core cooling capability. Phase 1 LOLA event evaluation focused on the operational response to explosion or fire. Prearranging for the environment of outside organizations, planning and the preparation activity and the developing procedure and the training for the event will be addressed by the COL applicant.

For the Phase 2 LOLA event we have -- it have -- we have to focus on mitigating on the event involving the spent fuel pool. APR1400 design adopt internal and external spent fuel pool makeup and the spray strategies.

For Phase 3 LOLA event evaluation focused on maintaining command and control, physical writer, and alternative mitigation strategy is develop to perform key safety functions.

MEMBER STETKAR: Now during the Subcommittee meeting we had some discussions about I'll call it coordination of the information in Chapter 19.3 and Chapter 19.4. We noted some discrepancies in for example different capacities, flow capacities that were listed for the same functional pumps in those two sections. And it was noted at that time that while 19.3 addresses guidance in one NEI document and 19.4 addresses guidance in a different NEI document, we have not yet reviewed Revision 1 of the DCD.

Have you done anything to resolve those discrepancies? I'm doing -- for example, I'm looking at 1,000 gpm flow capacity required for a certain pump versus I think it was 750 gpm. And different pump names and different inventories of pumps. Have you done anything to integrate that, because the -- I'll ask staff about this when they come up -- the pending regulations in 10 CFR; I always get it wrong, 50.155, I think it is, the mitigation strategies, indicates applicants and licensees should be integrating all of these strategies, not having the kind of diverse separate approach to life. So can you comment on that at all? And no is a possible answer.

MR. SISK: This is Rob Sisk, Westinghouse. I think the best we can say at this point is we're taking a lot of notes on what we are hearing over the ACRS Subcommittee meetings and the Full Committee meetings.

MEMBER STETKAR: Okay.

MR. SISK: We are -- as you've noted, we've issued Rev 1 of the DCD.

MEMBER STETKAR: Yes.

MR. SISK: There will be additional revisions as we close out confirmatory items and looking at things like consistency of language and such.

MEMBER STETKAR: Okay.

MR. SISK: So we hear it and we consider it as we go forward with the next revisions of the DCD.

MEMBER STETKAR: I personally would hope; and again I'll ask the staff about this because they're reviewing it, that by the time the final DCD is issued that there is more of a sense of and integration between 19.3 and 19.4.

MR. SISK: And we've captured that and we'll be looking into that.

MR. Y. KIM: Yes, the final section is about the aircraft impact assessment. Aircraft impact assessment of APR1400 was performed in accordance with Reg Guide 1.217 and NEI 07-13. The integrity of the containment and the spent fuel pool is maintained for all strikes to meet the AIA requirement. The heat removal assessments demonstrate that at least one division of core cooling is available for all strikes.

Regarding the key design enhancement, we have reinforcement and local strengthening measures are taken to protect critical hallways and penetration into containment. Physical separation of MCR and the remote shutdown room and on-site power resource such as EDJ and AEACEs are adopted in our design.

To prevent the fire and the shock damage due to aircraft impact, three hour late fire failure and the high PSI barrier are incorporate in our APR1400 design.

Let's move to next slide. Regarding the assessment result, first one is core cooling capability. Reactor cooling boundaries are not perforate and protect all internal safety system. And the integrity of a polar crane is maintained. And critical hallways and the containment penetration are physically protected. And the safety system in at least one quadrant will remain functional for all strike scenarios.

With respect to integrity of the spent fuel pool, spent fuel pool is protected from direct strike and the spent fuel pool wall and the support structure will survive secondary impact if external walls are perforate. And so spent fuel pool will survive secondary missile from above if external wall are perforated. And spent fuel pool will survive the roof collapse if the external walls above spent fuel pool are perforate.

So in conclusion, APR1400 meet 10 CFR Part 50.150. So we -- this AIA result shows that the cool -- core cooling capability and the integrity of the spent fuel pool. That is my end of presentation.

MEMBER STETKAR: Can I come back to those FLEX strategies, because I took a -- it's slide No. 27 --

MR. Y. KIM: Twenty-seven.

MEMBER STETKAR: -- where I asked about the coping time for the low power. I just pulled up that section of Revision 1 of the DCD. And indeed, that information is still in there. The Phase 1 coping time is listed as three hours. And as we become familiar -- in Section 19.3.2.3.1.3, for the record -- so that it still is included in the DCD as part of the strategy.

MR. Y. KIM: Yes, but we also provide the separate technical document through the NRC. So there was a -- could you help me?

MR. OH: Yes, for the -- when I checked for 19.3 for the shutdown strategy for -- I corrected my statement. I told that all the strategy is eliminated, and that's not correct. It's for -- yes, for some modification is done based on --

(Simultaneous speaking.)

MEMBER STETKAR: Yes, I -- so I see change bars in the margin and I can't read very quickly and speak and listen at the same time, but I did at least confirm that Phase 1, 0 to 3 hours; Phase 2, 3 to 72 hours; and Phase 3, after 72 hours is still listed explicitly in the DCD. As I mentioned, I personally an happy to see that. And I'm glad to see that it was not removed. Details of the changes we can review when we see Rev 1 of the DCD. I just wanted to get that on the record while we have you here to see if you wanted to comment further.

MR. OH: Yes, there's some modification has been done for that strategy according to the COL action item.

MEMBER STETKAR: Okay. Thank you.

MEMBER BALLINGER: Finished?

MR. SISK: This is Rob Sisk, Westinghouse again. And that does conclude our presentation on the Chapters 6, 12, 13, 14, 16, 17 and 19 for today, unless there are any other questions from the Committee.

MEMBER BALLINGER: Mr. Chairman, I would propose a --

CHAIRMAN BLEY: Yes, we will do a --

MEMBER BALLINGER: -- break until 10:15.

CHAIRMAN BLEY: We're in recess for 15 minutes until 10:17 by that clock.

(Whereupon, the above-entitled matter went off the record at 10:01 a.m. and resumed at 10:17

a.m.)

CHAIRMAN BLEY: Meeting will come back to order, please.

Dr. Ballinger, back to you.

MEMBER BALLINGER: Is the staff ready to roll? That's a local term.

(Laughter.)

MEMBER BALLINGER: Bill, it's up to you.

MR. WARD: Thank you. We have the staff presentation now. It's going to be done by the chapter project managers and myself, and we're going to defer to the tech staff and the audience for the tough questions. So that's why we have them all here.

So we're going to cover all the chapters. We're going to cover them by project manager. And I have a few early slides.

As you know we've completed all the SERs, so we're through Phase 2 with open items. All the SERs have been submitted to the Committee. We are in the process of completing Phase 3. We have one more Subcommittee meeting with Chapters 7 and 18 in a couple weeks.

As I said, we've issued all the SERs. We've had a Full Committee meeting for five of the chapters: 2, 5, 8, 10 and 11, and we're now covering essentially seven chapters. And we have more Full Committee meetings scheduled in the future with 3, 4, 9 and 15, and then later 7 and 18.

We have issued SERs with no issues for the -- three of the topical reports, and two more are still in process. And I know one of the topical reports is QAPD, which will come up later. Another one KHNP has already issued the accepted version of that. And they're working on a second one, so we're making progress on topical reports as well. And those are included in the future scheduling.

So the first presentation is on Chapter 13, which is conduct of operations. I will present that. It has seven sub-sections. For the most part Chapter 13 is the conduit for the applicant to pass on how they expect operations -- operational programs, etcetera, to be conducted and provide any information. So 13.1 provides for the organizational structure of the applicant and it requires that they develop the management and organizational structure, including design, construction, operating, and maintenance responsibilities. This includes qualification requirements such as education, training, and experience. There were COL items in this section. We reviewed the COL items. We had no issues or technical challenges in that section.

13.2 is specifically on training. Again, it provides a number of COL items. There were no technical challenges there. There was one issue where the COL item made the wrong reference and we asked and RAI and corrected that, so there were no major issues on 13.2.

13.3 is emergency planning. There were no technical challenges there. There's a lot more to review in 13.3 in the Technical Support Center size and location. Descriptions were acceptable. The interfaces between emergency planning and various other parts of the DCD were reviewed in accordance with the SRP and it was determined that they were acceptably discussed in the other sections of the DCD.

13.4 is the operational program implementation. This is all pretty much COL requirements. We have a SECY paper, an SRM that give the guidance for the staff as to how to approach this. So this was reviewed in accordance with that paper. The COL item was found acceptable.

13.5 is plant procedures. Again, it contains COL items. The applicant briefly described admin and operating procedures for all operational modes and is scheduled for preparing the procedures.

This is also where the applicant, DCD applicant provides Emergency Operating Guidelines.

We reviewed the COL items. We found that they were adequate and acceptable. Some of those are -- were being edited. There was some question about some wording. So we're waiting to confirm the confirmatory actions on those.

The Emergency Operating Guideline were found to be technically adequate and acceptable for the COL to use as a basis for procedure development. And we also reviewed a technical report, "Best Estimate Analyses for Operational Transients and Accidents for APR1400 EOGs," and found that it was acceptable. And as KHNP discussed earlier, 13.6 is physical security and 13.7 is fitness for duty, and those were not presented to ACRS, but they've been reviewed.

In summary, all the regulatory requirements for Chapter 13 have been satisfied.

And if there are any questions, I'll attempt to answer them for you.

(No audible response.)

MR. WARD: Okay. Next is Chapter 14 -- or I'm sorry, 17. Seventeen is quality assurance and reliability assurance.

I'll be presenting that. Sections are provided. There was a Quality Assurance program description, QAPD, topical report submitted at the very beginning of the project, actually in preapplication. And that was reviewed and approved, and it's up to I believe Revision 5 now. That report provides the requirements for the QA sections, which are 17.1, 2 and 3 and 17.5. That report and the review of those sections shows that all the regulatory requirements of Appendix B to Part 50 have been satisfied. There are some COL items in Sections 17.2, 17.3 and 17.5, and those were found acceptable.

QA inspections have been performed. One was performed a year ago; another one was completed recently. And you'll have to look for the reports for those as that's not really our realm to discuss.

Staff did review the D-RAP, Design Reliability Assurance Program, ITAAC and compared the list with PRA results. And that program was found to be insufficient primarily because the applicant used old guidance. And an RAI was issued and new guidance was -- is being used for them to update their list. And when that's completed we'll be able to complete the review of that list.

MEMBER STETKAR: Bill, but that list will change after they change the PRA, or it could change after they change the PRA again.

MR. WARD: That is correct.

MEMBER STETKAR: Okay.

MR. WARD: And there will be more discussion in 19.

The maintenance rule section and the COL item were reviewed and found to be acceptable.

That's all we have on Chapter 17. Any questions on 17?

(No audible response.)

MR. WARD: Okay. Chapter 6.

MS. UMANA: Good morning. I'm Jessica Umana. I'm the project manager for Chapter 6, which is engineered safety features. The staff's areas of review for Chapter 6 covered materials for engineered safety features, containment systems, the safety injection system, habitability systems, fission product removal and control systems, in-service inspections of Class 2 and 3 components, and in-containment -- I believe that should be in-containment refueling water storage system.

Most of the regulatory requirements for Chapter 6 have been satisfied.

We do have a few remaining issues which

I've listed there and I'm going to briefly provide you some information on the items that are listed there.

To start with, the staff is working and communicating frequently with the applicant to develop a path forward and bring closure to these items.

In the Subcommittee there was a question about the swing panels. The staff has issued and RAI requesting the applicant to provide information describing whether or not the IRWST swing panels are required to function properly and need to be included in the tech specs in order to ensure continued operability of the ECCS.

Moving on combustible gas control in containment, which is Section 6.25 in the DCD. There was a question that Dr. Rempe had from the Subcommittee, and I believe the question is what is the basis for the staff's satisfaction with MELCOR containment nodalization.

The staff provided a response to Dr. Rempe on May 8th. Containment modeling is addressed in many -- contained in MELCOR reports where various nodalization schemes have been shown to produce results consistent with large experimental programs such as NUPEC, HDR, and CVTR. These nodalization schemes appropriately reflected both the openness and complexity of the internal geometry of the specific containments.

APR1400 has a large dry open containment. The APR1400 nodalization both in the applicant's MAAP 4 and the staff's MELCOR model reflects openness of the containment design as well as the locations and elevations of the PARs and hydrogen igniters.

In summary, because the nodalization scheme used to evaluate the mixing of containment for APR1400 is sufficiently similar to the nodalization for similar containment designs previously approved. The staff's confirmatory calculation included more nodes in the upper containment dome region than the applicant's, and both the applicant and the staff's results were within reasonable agreement with the APR1400 results. The staff concluded that further sensitivity was not necessary.

Moving onto boron recycling. The staff is engaged with -- in discussions with the applicant. The applicant will provide a discussion addressing how the APR1400 design will control boron-10 concentration in the IRWST, the safety injection tanks and the spent fuel pool since APR1400 has the capability of recycling boron. Currently the applicant has proposed adding a tech spec or surveillance requirement for boron-10 control, however, we're still in the draft stages and don't have much more information to provide on that front.

For the remaining issues the -- I'm sorry, the last issue, which is containment structure and mass and energy release, which I believe is Section 6.211 and 6.213 and 6.214 in the DCD, nothing has changed since the Subcommittee. We are also engaged in back and forth exchange with the applicant to bring resolution to the open items for those sections.

And that's it for Chapter 6.

MEMBER POWERS: In the applicant's presentation he assures us that in the post-accident environment the IRWST water will remain at pH 7 or higher throughout a 30-day period. Were you able to confirm that conclusion, and how did you go about doing that?

MS. UMANA: This is for the swing panels. I believe you had the question on the swing panels last time or --

VICE CHAIR CORRADINI: I'm the swing panel person. He's the pH person.

MS. UMANA: Okay.

(Laughter.)

VICE CHAIR CORRADINI: Sorry.

MEMBER POWERS: The swinger and the stable --

(Laughter.)

MEMBER POWERS: -- sane individual.

MEMBER STETKAR: The old acidic guy.

MS. UMANA: The -- I don't think I have -- I know who this question is for and I don't think he's here. Andrew Yeshnik's not in the crowd, is he? Or was it Greg Makar? No. I'll have to take your question back and provide you --

MEMBER POWERS: Yes, that's fine.

MS. UMANA: -- a response at a later date.

MEMBER POWERS: That's fine. I mean, just point me in the right direction.

MS. UMANA: Okay.

MEMBER POWERS: Thank you a lot.

MS. UMANA: I didn't anticipate that question. How's that?

(Laughter.)

MEMBER POWERS: Well, similarly in this; I'm in Chapter 6, and kind of in Chapter 2 they look at the dispersion of radioactivity toward the main control room. How did you look at that?

(No audible response.)

MEMBER POWERS: It's a complex analysis

which I suspect one has to do fairly simply because I don't think you can do it phenomenologically correct.

MR. WARD: You're asking about the dispersion of radioactivity in the main control room?

MEMBER POWERS: Yes, around the intakes and things like that.

MR. WARD: In any particular condition, or just --

MEMBER POWERS: Post-accident.

MR. WARD: Oh, post-accident?

MEMBER POWERS: Yes.

MR. STECKEL: This is Jim Steckel. We do have a discussion of that in Chapter 15 with regard to the opening and closing of the vents. And we did have several questions to iron out with KHNP.

MEMBER STETKAR: Jessica, in -- I was just reviewing my notes from the Subcommittee meeting and we had some discussions about whether or not the applicant includes credit for what's euphemistically known as containment accident pressure. I just pulled up Rev 1 of Chapter 6 of the DCD and I note that they do consistently with currently operating plants -- that they account for pressure equivalent to the saturation pressure for the temperature of the IRWST water. So for example, if the IRWST is at 110 degrees

C, they'll account for that amount of overpressure in the containment to maintain net positive suction head for the pumps. Has the staff reviewed any of that, those analyses?

MS. UMANA: I do not know.

MEMBER STETKAR: Okay.

MS. UMANA: And again, I --

MEMBER STETKAR: There was some -- my notes during the Subcommittee meeting, there was --

MS. UMANA: I recall the discussion, but --

MEMBER STETKAR: -- some discussion about whether or not the pumps would survive if the containment were open to the environment. And there seemed to be some people saying yes they would. Other people -- but Rev 1 of the DCD seems to clearly indicate -- it says for higher IRWST sump fluid temperatures the containment pressure is assumed to equal the saturation pressure corresponding to the sump water temperature.

MS. UMANA: I think we may have an answer for you, right?

MR. TRAVIS: So this Boyce Travis, the NRC staff. The treatment we discussed in the Subcommittee meeting is consistent with the description in Rev 1 of the DCD. Those are just RAI responses to update. The analyses do assume that the pressure credit for NPSH assumed in the vapor pressure cancels out with the saturation pressure of the fluid, as you stated.

In the event that -- so beyond the design-basis, an event that -- a loss of containment pressure would result in flashing of the steam and the -- if you look at our table, the margin they have available in the post-accident condition versus the condition that you're describing, that Dr. Corradini has the slide available that I presented at the Subcommittee meeting, they -- we didn't review that case specifically because that's outside the design-basis of the plant.

I'll state that based on the information they provided the expectation is the same amount of margin would be available in the case with atmospheric containment pressure. If you open it instantaneously versus the case -- the post-accident case where they're crediting the vapor pressure up to the saturation temperature of the fluid in the IRWST.

MEMBER STETKAR: Well, it can't be the same margin available. The question is is adequate margin available?

MR. TRAVIS: So once the sump temperature reaches 212, based on how their analyses are credited, the same margin is available from that point on because they're crediting only the saturation pressure associated with the temperature of the IRWST fluid. They don't credit additional pressure in containment beyond the saturation temperature of the fluid.

MEMBER STETKAR: Beyond the pressure for the saturation temperature?

MR. TRAVIS: That's correct.

MEMBER STETKAR: So if the fluid was at 250 degrees Fahrenheit, the saturation pressure is -- I don't know what it is. Thermal-hydraulic people can tell me what it is.

MR. TRAVIS: Call it 35 psi, something like that.

MEMBER STETKAR: Yes. They account for that?

MR. TRAVIS: That's correct.

MEMBER STETKAR: Okay.

MR. TRAVIS: But no higher than that.

MEMBER STETKAR: But no higher than that. Right. I'm not arguing about no higher than that. On the other hand, if I make 35 psi -- I don't know whether you want to talk atmosphere or gauge, because I don't remember -- if I make it 0 psi gauge, that will reduce their margin if the temperature is 250 degrees Fahrenheit.

MEMBER MARCH-LEUBA: But in that case; just to help you, in that case you can not maintain 250 --

MEMBER STETKAR: Correct.

MEMBER MARCH-LEUBA: -- because you will flash and cool off the --

(Simultaneous speaking.)

MEMBER STETKAR: That's right, but you'll flash first at the eye of the pump and it will cavitate.

MEMBER MARCH-LEUBA: It's a complicated calculation.

MEMBER STETKAR: It is a complicated calculation. That's why pump manufacturers don't like you to be in that -- I'm just trying to make sure that I understand what is in the DCD, and I think I do now.

MR. TRAVIS: Okay.

MEMBER STETKAR: Thanks.

MR. TRAVIS: Do you require any further clarification? Should I stay up here I guess is the question I'm asking.

VICE CHAIR CORRADINI: Well, since you brought up your presentation, remind me, since I remember you explained it; now I can't remember, I didn't write notes myself, the duration is actually fairly short with holdup. So remind me what with holdup means again?

MR. TRAVIS: So that analysis is not specific to the APR1400. That --

VICE CHAIR CORRADINI: I understand that.

MR. TRAVIS: Okay.

VICE CHAIR CORRADINI: It was -- but you wanted to explain to us --

MR. TRAVIS: Sure. So the analyses in the DCD do not account for holdup of water in other locations. It's a single containment node, so all of the hot water goes instantly to the sump. In reality some of that hot water is going to be held up in for instance the holdup volume tank, various dead end compartments in the steam generator rooms. And so that hot water is going to act to reduce the sump temperature and therefore shorten the duration that the sump is above 212 degrees Fahrenheit.

VICE CHAIR CORRADINI: Thank you.

MR. TRAVIS: No problem.

MEMBER REMPE: So are you -- do you have another comment?

MR. TRAVIS: No problem.

MEMBER REMPE: Okay. No, I was talking to Mike Corradini. His hand was up and I thought he wanted to talk more. But it's gone now.

I wanted to thank you for your response back on the hydrogen nodalization.

MS. UMANA: Okay.

MEMBER REMPE: And I appreciate it and I don't need any further clarification, but I would like to mention that there is an ongoing effort to get -- I may be saying it incorrectly, but the HYMERES facility where they're getting CFD quality experimental data to reevaluate this issue if the nodalization is correct. And I believe research has started in an effort with Caltech to try and evaluate that. So might want to think about it in the future, but I don't know the timing. And so again, I think it's fine for what we're doing today. We don't have additional knowledge yet, but there might be in the future.

MS. UMANA: You said HYMERES?

MEMBER REMPE: It's H-Y-M-E-R-E-S. I may be mispronouncing it. Do you know how to pronounce it better?

MS. GRADY: This is Anne-Marie Grady and I'm the staff that provided the response. And thank you for that reference to HYMERES.

MEMBER REMPE: Okay. Thank you.

MS. GRADY: And I will look into that.

MEMBER REMPE: Okay. Thanks.

MR. WARD: Any other questions on Chapter 6 6?

MS. UMANA: I'm sorry, do you want to go back to Chapter 6 and attempt to address --

MR. McCOPPIN: I think you asked the question -- this is Mike McCoppin -- the question about the dispersion in the main control room. Michelle Hart's here, and she might be able to address that question.

(Pause.)

MS. HART: I just got here.

MR. McCOPPIN: Yes, Michelle just got here. She'd like the question repeated, if she could.

MEMBER POWERS: Oh, okay. Michelle, what I'm interested in is how we calculate the dispersion of radioactivity around the main control room in the Technical Support Center.

MS. HART: Are you talking about within the control room?

MEMBER POWERS: No, no.

MS. HART: To the control room?

MEMBER POWERS: To them.

MS. HART: We used the RCON 96 computer code, and it is based on hourly data for a year, much like we do for the off-site analysis, but it does include building wake effect. And if there's a lot of heat behind it, it does include lofting. But this plan there's not that --

MEMBER POWERS: Didn't find any --

MS. HART: And there's no effect from being released from a stack either.

MEMBER POWERS: Oh, okay. As long as I've got you here, in their assertion of pH 7 what kinds of things did we look at to make sure that they maintained a pH of 7 in the IRWST for 30 days?

MS. HART: Well, I do have to admit I did not look at that analysis. That was somebody from the Chemical Engineering Branch.

MEMBER POWERS: Oh, okay. Okay. Fine.

MS. UMANA: Any other questions related to Chapter 6?

(No audible response.)

MR. WARD: Okay. We'll move onto to Chapter 16.

MS. UMANA: I also have Chapter 16, which is the generic tech specs. Yes, it's my favorite.

That's quasi-true.

(Laughter.)

MS. UMANA: The tech specs are based on combustion engineering standard tech specs, NUREG-1432, Revision 40. The difference between the combustion engineering tech specs and the generic tech specs are based on design variations unique to APR1400.

A review of the generic tech specs has been conducted resulting the safety evaluation chapter that includes open items in the following areas: Adequacy of requirements during reduced RCS inventory, requirements to prevent inadvertent reactor coolant boron dilution, AFW required actions and completion times appropriate to the APR1400 design, CRHS required actions and surveillances, accident monitoring instrumentation requirements, and application of LCO criteria, TSTF disposition and COL action items.

Regarding the application of LCO selection criteria, the applicant did not use a systemic process of evaluating the safety analysis and SSCs against the criteria, but relied on evaluating applicability of combustion engineering standard tech spec's LCOs to APR1400 SSCs in applying engineering judgment.

The staff requested the applicant to

consider additional measures to ensure the criteria is satisfied; that is, that the tech specs include LCOs for all SSCs, parameters and initial conditions that satisfy their criteria.

The open items are being discussed. We have continuous back and forth exchange with the applicant in coordination with other branches for closure to the open items related to Chapter 16.

One thing I did want to mention before concluding this presentation for Chapter 16 -- I wanted to talk about requirements for mitigating shutdown risk. They have been included and go beyond requirements in the standard tech specs.

In the Subcommittee presentation given in March of this year there was a question about the role of risk insights into the development of the tech specs. It's correct to say that the tech specs are not risk-informed and that significant risk-informed initiatives have not been incorporated. However, LCO 3.0.8, which is based on TSTF Traveler 372 and relates to inoperable snuffers and LCO 3.0.9, which is based on TSTF Traveler 427 and relates to inoperable barriers, require risk insights for adoption. These risks are presented in the form of generic risk evaluations which are included with the travelers.

The applicant provided an equivalent evaluation for LCO 3.0.8 and is considering providing an evaluation for LCO 3.0.9.

And finally, pending resolution of the open items verification of compliance with the regulations is not yet quite complete.

MEMBER STETKAR: I would just --

MS. UMANA: Any questions?

MEMBER STETKAR: No.

MS. UMANA: I know you have questions.

MEMBER STETKAR: No, I don't actually. I was just going to make a statement. That being said, there's this -- for the benefit of the other Committee members, there's this notion of are the tech specs risk-informed? And they're not risk-informed according to the process related connotation of risk-informed.

However, as Jessica noted, the tech specs do have specific LCOs in them in particular for shutdown, which she mentioned, that evolved from information that came out of the risk assessment. So the applicant used the risk assessment and insights from the risk assessment to establish conditions in their tech specs despite the fact that they are not, quote/unquote, "risk-informing" their completion times or their surveillance intervals. I hope that's accurate.

MS. UMANA: If there are no other questions for Chapter 16, I'm going to turn it over to the next presenter.

MR. WARD: The next chapter will be Chapter 12.

MR. TESFAYE: Good morning. My name is Getachew Tesfaye. I'm the project manager for Chapters 12 and 14. I have a couple of staff member here to support me if there are any questions.

Chapter 12, radiation protection. The Subcommittee presentation was on February 24. We looked -- the staff looked at various sections. The first one is ensuring that occupational radiation is ALARA. Section 12.2, radiation source. Section 12.3 and 12.4, radiation protection design features, including dose assessments. And the last section is Operational Radiation Protection Program.

For Section 12.1 and 12.5 all the regulatory requirements have been met. There are no open items. For Sections 12.2, 12.3 and 12. 4 and -- I repeated 12.5 here again, so I apologize for that. The regulatory requirements for these sections have been satisfied with the exception of five open items in Sections 12.2 and nine open items in Section 12.3 and 12.4.

So the staff has been working on these open items. There are no show stoppers. We have to complete our Phase 4 review of Chapter 12 soon and be able to present the completed SER with no open items in Phase 5.

That's all I have. Any questions for Chapter 12?

(No audible response.)

MR. WARD: The next chapter is Chapter 14.

MR. TESFAYE: Chapter 14 is kind of complicated. We've done 14.1 and 14.2 in Phase 2, and we deferred 14.3, the ITAAC for Phases 4 and 5. In Section 14.1 of course there's no open items. In Section 4.2 staff looked at the four phases: pre-operational testing, fuel loading and post-core hot functional tests, initial criticality and low-power physics tests, and power ascension tests. The number of tests in each category may increase as a result of response to open items.

Currently for Phase 2 we have 16 open items in Section 14.2. And again, 14.3 is -- the whole section is an open item that will be addressed in Phase 4.

That's all I have for Chapter 14. Any questions?

MEMBER KIRCHNER: Yes. So the open items included things like steam generator blowdown, hydrogen mitigation, liquid and gaseous waste management, etcetera. Is this because the COLA will address those items, or why are they -- I'm just curious. Why did they remain open at this phase?

MR. TESFAYE: Well, they're open because the applicant needs to provide more information, whether this is going to be a COLA item or the laboratory test to this chapter. So it's pending the applicant's response.

MR. WARD: Any other questions?

(No audible response.)

MR. WARD: Was that response satisfactory?

MEMBER KIRCHNER: Yes, thank you.

MR. WARD: Okay. All right. We'll move onto Chapter 19. This is going to be 19, 19.3, 19.4 and 19.5. And it's Jim Steckel.

MR. STECKEL: Thank you.

CHAIRMAN BLEY: Why don't I sneak in the question I asked the applicant earlier?

From what they told us this time around when we get to the COLA stage they're saying the COLA

PRA will need to do a seismic PRA and things associated with local water conditions and events. Is that your understanding as well? I mean, the past other applicants have done kind of a surrogate for those things and then only required at the COLA stage that people show they're better than the surrogate. We don't have that this time, so it sounds like they're doing it what I would call the right way this time.

MR. STECKEL: I'm going to -- this is Jim Steckel. I'm going to defer to one of the reviewers.

Hanh, can you answer that?

MR. PHAN: Hello, Hanh Phan, the lead reviewer for PRA.

So far the applicant provides us the PRA page SMA. And according to the staff guidance that is acceptable for the DC applicant. With that we would not ask them to further perform any analysis or address any CDF or error now at this point. However, there is a clear COL information items asking the COL applicant to perform a full-scale PRA, seismic PRA when they have the site-specific information.

Knowing that, because during the DC phase there are some information not available, so the risk insights gained from any further assessment would not contribute much to the staff positions regarding their application.

CHAIRMAN BLEY: I can't say that I understood that answer. I don't. And it -- the whole process doesn't seem consistent with what we've done elsewhere. So the applicant said people doing the COL will have to do a -- those site-specific analyses. And it sounds like you said no they don't.

MR. PHAN: Yes. Yes, they will do.

CHAIRMAN BLEY: That's what you said?

MR. PHAN: It's the other applicants --

(Simultaneous speaking.)

CHAIRMAN BLEY: Okay. I misunderstood what you said.

MEMBER STETKAR: Let me, because I'm confused again. Let me ask you a very specific yes/no question.

Will the COL applicant's submittal to the NRC for the COL review contain a risk assessment that includes quantification of the seismic risk? That's a yes or no question.

MR. PHAN: No.

MEMBER STETKAR: Okay. That is not what I understood from everyone else's discussion. When will that risk assessment that quantifies the risk from seismic events be expected from someone who will eventually operate one of these plants?

MR. PHAN: Before the 103 filing.

MEMBER STETKAR: Well, since --

PARTICIPANT: Before fuel load.

MEMBER STETKAR: Thank you for putting that into English. Before fuel load?

MR. PHAN: Yes.

MEMBER STETKAR: Okay. Thank you. That's on the record now from the staff.

MR. PHAN: May I be clear by saying a little bit more? In the COL phase there are RCOL applicants and SCOL applicants. For the R, the reference COLs, they can -- under Part 52 they can submit their application early using DC information. By that, they can use the information in the DCD Chapter 19 to justify the risk, but however they have to do more by performing the PRA, seismic PRA to identify the risk.

MEMBER STETKAR: I'm sorry, you said the words "seismic PRA," so I will ask you a specific question. Will they submit a risk assessment at the COL? Since you want to differentiate between RCOL and SCOL, will an RCOL applicant be expected to submit a risk assessment that quantifies the risk from seismic events? Again, yes or no?

MR. PHAN: Not the quantification risk.

MEMBER STETKAR: Thank you. That's what I consider risk. Anything else is not a risk assessment.

MR. PHAN: Thank you.

CHAIRMAN BLEY: I'd offer two things, and I hope the staff thinks about this long and hard: The previous design certs included a section in Chapter 19 that essentially bracketed these environmental events and showed they were I'll just say okay under those specific conditions, which then a COL applicant had to show that they met that list of conditions to avoid doing -- including those in their COL PRA. That isn't true this time, so I think you ought to think about this a bit.

And secondly, there ought to be no distinction between an RCOLA and SCOLA because every site's different. Enough said for now.

MR. STECKEL: Is there another question?

(No audible response.)

MR. STECKEL: If not --

CHAIRMAN BLEY: From me? No.

MR. STECKEL: Okay. We'll move on.

Chapter 19 contains the four main sections shown there. The probabilistic risk assessment and severe accident sections take up 19.1, 19.2. 19.3 is beyond-design-basis external events; 19.4, loss of large areas; and 19.5, aircraft impact assessment.

Next slide, please? So concerning PRA the areas of review are considering the scope, techniques, level of details and technical adequacy: Internal events Levels 1 and 2, at-power and during low power and shutdown; internal fire Levels 1 and 2, at-power and during low power and shutdown; internal floods Levels 1 and 2, at power and during low power and shutdown; PRA-based seismic margin assessment; and other external events. And we also did key assumptions and risk insights; PRA maintenance and update; uses and applications of APR1400 design-specific PRA.

Next slide, please? So the main focus areas here are: The PRA model conversion from SAREX to CAFTA, which is ongoing; PRA peer review findings and resolution; digital I&C modeling; seismic assessment during low power and shutdown; PRA update; consistency between the DCD and supporting information such as the PRA notebooks; sensitivity studies, risk insights, update of PRA input to other DCD chapters and programs; and RAI reconciliation.

If there are no questions on 19.1, we'll move to 19.2, severe accident evaluation.

MEMBER STETKAR: The fifth bullet under there says PRA update. What are you going to do about that? I know you've done audits for the I'll call the Rev 0, Rev 1 version of the PRA. Are you planning to do additional audits after they change the models and have evaluation?

MR. STECKEL: We -- this is an area of discussion we're having right now, and an area where we'll need to engage again with ACRS to determine their agreeability to have additional review.

MEMBER STETKAR: Okay. So it's still a work in progress?

MR. STECKEL: It's still a work in progress --

MEMBER STETKAR: Thank you.

MR. STECKEL: -- at this point.

And, Hanh, did you need to mention anything more?

MR. PHAN: Yes, I have further clarification regarding APR1400 PRA models.

Right now the applicants, they do have three models. The original model in the original submittals using the SAREX software. During the review they told us that they convert their models into another software, CAFTA. During that conversions they incorporate the -- some of the staff findings, some of the peer review findings and their self-assessment, the type of methods in their original models.

With that, last year they present to us the quantification of the CAFTA models and they told us that complete this, but we have not seen that model yet. We only saw their presentation. Right now they told us that they finished with that conversion and they are working on the update.

The update means that they're going to incorporate all the findings from the peer review, from the staff RAIs and any comments from that ACRS. That model would not be complete until September, so staff will have another reviews of that latest PRA model in the future, near future.

MEMBER BALLINGER: Now I'm confused again. And that is I heard the words "further engagement with the ACRS." And that is, at least the way I interpret those words, inconsistent with what Bill has said.

MR. WARD: I'll answer that.

Yes, I was checking with staff in the break and there is an interest in providing in a meeting of some sort some information prior to the actual Phase 5 Subcommittee, Full Committee meetings. Exactly what form that would take or what -- how much would be provided or when -- well, what's provided is going to depend on when, but there is an interest in some sort of meeting to provide you some advance information before we get to that point in Phase 5.

MEMBER BALLINGER: Okay. So that's a change?

MR. WARD: It has recently been proposed from staff, yes.

MEMBER STETKAR: Sounds like we're -- basically it's a work in progress at this snapshot in time.

MR. WARD: I think the goal is to provide you as much information as we can because we don't want to have surprises. Nobody does. But it's really hard to manage it with the schedule and the timing of the information coming in.

MR. PHAN: Hello, this is Hanh Phan again. Base on the comments from the ACRS the staff proposed that upon ACRS approvals that in Phase 5 we're going to have two presentations. One in the early of Phase 5 and one at the end of Phase 5. The early presentation would be sent to the Subcommittees on PRA regarding the staff observation on their latest PRA information. And then at the end of Phase 5 we will come back and present to you the staff accept the evaluation report without open items and including any comments, any feedback from the Subcommittee, ACRS Subcommittee on their PRA models. But that upon the ACRS approval.

MEMBER BALLINGER: I guess this is definitely a work in progress. So we'll have to deal with that, not here.

MR. WARD: I agree.

MEMBER BALLINGER: Not today, anyway.

MR. WARD: I agree.

MEMBER SUNSERI: And, Bill, that proposal that's been laid out, I presume that does not include at this time engagement with KHNP or any of their PRA folks. Is that correct?

MR. WARD: Not as I understand it.

MR. STECKEL: It could change. I think we can move onto 19.2, severe accident evaluation. These were the areas of review: accident prevention; accident mitigation, MELCOR confirmatory analysis; and containment performance capability.

An area that is still ongoing and is a strong focus area is the MAAP analysis of mid-loop accidents. There are -- there's calculations that we're doing, confirmatory calculations. And we're still working with KHNP to complete those. And the structural analysis of ex-vessel steam explosion --

(Simultaneous speaking.)

VICE CHAIR CORRADINI: So, Jim, can you explain the last bullet? I'm not sure what that means.

MR. STECKEL: I am going to get an expert so that I don't --

(Laughter.)

MR. STECKEL: -- misstate a thing.

So if the reviewer is available on this? Hanry? Hanry Wagage.

MR. WAGAGE: My name is Hanry Wagage. Applicant has presented results from --

MR. STECKEL: I think you need to get closer, Hanry. Kiss that mic, so to speak.

(Laughter.)

MR. WAGAGE: Hi, this is Henry Wagage. The applicant has presented results of structure analyses performed for preference plant, similar plant. So we have been reviewing these result. We were -- we had some questions. For example, how the applicant attenuated the pressure calculated from steam explosion, how they attenuated the pressure and applied on the tops of walls. That's one example.

So we had questions like that that the applicant has to address those questions. There's an open item because of that, how the applicant performed structurally.

VICE CHAIR CORRADINI: Okay. So it's -- the analysis is underway that you get to review?

MR. WAGAGE: That's right.

VICE CHAIR CORRADINI: Okay. All right. Thank you.

MEMBER REMPE: So I'd like to also chime in about some of this. The Subcommittee meeting we learned about the document the staff had prepared with the MELCOR confirmatory calculations. And so after that we had the opportunity to review it and we discussed some of those calculations with the staff. This has been an accelerated schedule for review. At the time the staff did those calculations, they had some differences in their assumptions regarding the plant. And maybe they didn't pick the sequence of most interest.

We've heard today earlier that there may be some differences in the risk assessment results. We discussed about the fact that the consequential steam generator tube rupture effects might be different when one looks at the natural circulation for a CE plant-specific configuration based on CFD for a CFD -- a CE plant versus a Westinghouse plant that was assumed.

It seems to me that the staff may want to look at the results from the revised risk assessment. They may even want to say I want to try and run a different case for a confirmatory calculation. Those kind of issues, are they still on the table or are we going to at least -- if they don't run additional analyses are we going to have some confidence that we believe that the confirmatory MELCOR calculations that were performed give us confidence about what may be very important with respect to the risk assessment?

MR. STECKEL: Just let me mention before Jason begins that this was a rather long RAI with many questions associated with it. Not all of them were specific. Many of them were we've noticed this in this limited number of things we've looked at. These things needs to be looked for in many other areas. And so that is still an ongoing process, but we'll get further clarification from Jason Schaperow.

MR. SCHAPEROW: Yes, this is Jason Schaperow of the staff. With regard to the steam generator tube rupture observation, as you mention, Dr. Rempe, the latest NRC report on that shows the likelihood of steam generator tube rupture for a CE plant of roughly 0.25 given a core melt accident for a CE plant, whereas for a Westinghouse plant it's about a factor of 10 lower.

The MELCOR confirmatory calculations that we're doing won't really shed any light on that because the analysis that was done for steam generator tube rupture is a combination of a materials evaluation of the tubes and the MELCOR calculations. And we're not going to be redoing that for this particular plant specifically.

With regard to your question about the tube rupture probability that was previously thought to be based on NUREG-1570, the numbers in there -- if I recollect correctly, were for Westinghouse-style plants were on the order of 0.25. So my understanding is that basically the numbers stay kind of the same from back in 1998 to now for -- except for the Westinghouse numbers got smaller. The original analysis ones for Westinghouse --

MEMBER REMPE: It would be good to see that --

(Simultaneous speaking.)

MR. SCHAPEROW -- 0.25.

MEMBER REMPE: I guess I'd like --

MR. SCHAPEROW: That's my understanding of it.

MEMBER REMPE: -- that because again, I thought the CFD analyses that were done for the Westinghouse plant -- let's see that documented and let me explore that more, because that would be good to see.

But the other question is for the analyses where you had different decay heats and things like that, there may be some differences in geometry. Do you feel like you'll have confidence in what they produce finally?

MR. SCHAPEROW: So for the at-power scenarios we believe we have reasonable agreement with the MAAP results. For the shutdown scenarios, the mid-loop calculation particularly, we've identified some issues we think with their calculations. And they're redoing those calculations now and they're redoing the documentation. So we're waiting to see what the new documentation says about their updated MAAP calculations. And theoretically that would also be reflected in our updated PRA.

MEMBER REMPE: And you don't think you

need to do any sort of revised MELCOR analyses?

MR. SCHAPEROW: We are in a position that we could do some more, but again I would wait to see what they come up with before we start spending effort to redoing any MELCOR calculations at this time.

MEMBER REMPE: So if you do that, if it -- you make that decision to do that, which again it may not be necessary; no one wants to waste money, there were some calculations done in a separate SORCA evaluation for Korea. And so my understanding is there might be a better model to start with for the MELCOR calculations. Have you -- are you aware of what I'm talking about --

MR. SCHAPEROW: I --

MEMBER REMPE: -- that there might be such a model --

MR. SCHAPEROW: I have --

MEMBER REMPE: -- that would start with a plant that might be --

MR. SCHAPEROW: Yes, I've heard of --

MEMBER REMPE: You might want to compare those plant models, at least.

MR. SCHAPEROW: I'll need to check with Research on that. I think they may know more about that than I.

MEMBER REMPE: Okay. It might be good to do that.

Then the last thing I'd like to talk about is this thing about in-vessel retention and this optional system, or option, and what's -- where -- how much was the staff reviewing it? Because if you don't take credit for that system, there's a heat transfer coefficient from the vessel. What was assumed? I mean, it seems like that you've got this system in there and you said, well, okay, we didn't take credit for it, so we're going to let the vessel fail. But then in reality are there some unintended effects by making that observation we're not going to evaluate it?

MR. WAGAGE: Actually -- this is Hanry Wagage from the staff. The applicant did not take credit for this system for steam explosions. They did not take credit. Because of that there is no reason for the staff to look at how good the heat transfer coefficient, how well is going to cool.

Staff's concern was that it was different. If the vessel is -- if the cavity is flooded covering part of the vessel, if by using the system, if the melt freezes and -- at the bottom and vessel is failing at the site on the top by metallic layer, then much larger melt core would occur. That's the concern staff had, not that how well the system worked, because the applicant did not take credit for the system.

VICE CHAIR CORRADINI: So, Hanry, since you brought it up; it's your fault, I thought APR1000 went through all of this in -- back in -- 20 years ago. So staff was okay with it then. What changed? All of this in terms of the physics and the situation geometrically are quite similar.

MR. WAGAGE: Thanks for the question.

VICE CHAIR CORRADINI: No problem.

MR. WAGAGE: APR1000 --

(Laughter.)

MR. WAGAGE: -- during the review of APR1000 staff had the same concern. Then applicant for APR1000 argued that kind of much larger melt core would not occur.

VICE CHAIR CORRADINI: Right.

MR. WAGAGE: However, staff raised an issue that there are so much uncertainty. Then the applicant went ahead and did structural analyses to show that with much larger melt core there will be much larger energy generation by steam explosion. However, only the cavity would fail. But the content would stay intact.

VICE CHAIR CORRADINI: Okay. So your --

MR. WAGAGE: So that's how it is you close for APR1000.

VICE CHAIR CORRADINI: So your point is that analysis is yet to be done, but you don't need to do it simply because applicant is not offering this as the base design? This is -- we'll call it an optional addition in the design?

MR. WAGAGE: That's right.

VICE CHAIR CORRADINI: Okay. Fine.

MEMBER REMPE: But before you went with the APR1000 folks, you first of all had a bunch of questions about how robust is this cavity cooling system. Will it withstand the event? Basically you made them reinforce it so that it could always have water come in and the steam could be generated externally. Those kind of calculations or evaluations or will not be done if you say, okay, we're not going to take credit for it. It doesn't matter.

Furthermore, someone has in their MAAP evaluation made certain assumptions about the heat transfer from the vessel. Now you've got an isolated volume. If you don't take credit for it, what did you assume -- I mean, you have this insulation, so you have a gap. And so you have less heat coming out of the vessel than you would if you did not have that system there. There is that system there that could actually fall into the vessel, that could even make the situation worse.

And so those kind of discussions are not occurring because we're not taking credit for it. And so maybe that's fine. You throw it off to the COL applicant, but then somebody, if they ever want to take credit for it, needs to realize that it's going to be a lot more evaluations that have to be performed. And that's what I'm trying to argue for with all these questions today.

VICE CHAIR CORRADINI: They didn't want air-conditioning so they're not going to evaluate air-conditioning.

MEMBER REMPE: Yes. But then -- but there are some adverse effects. And so, and that ought to be documented in the staff draft SE or in the -- in the whatever that we -- we're assuming this didn't exist and they need to verify that the applicant truly did assume it didn't exist and considered the fact that there's something in there that they're not evaluating, and they properly considered that. And then say if you ever want to come in and take that option, you got to do a lot of analyses.

MR. WAGAGE: I agree, because the applicant needs to take credit for this system. Staff heard that applicant has to show how good the system works. Right at this stage there is very little information provided in the DCD. In one place it's mentioned there are insulations around the vessel at the bottom, that there is no -- not much information. Just one sentence.

So how was -- staff didn't have any concern about that, because there is no reason how well the system works. But opposite if the system does not work. If the system is operating and unintended consequence happens, like larger melt core. That's what the staff was focused on.

MEMBER REMPE: Yes, okay. So I think we're communicating, and it's just another issue. If we have another interaction before -- what was it called, Rev 5 or whatever --

PARTICIPANT: Phase 5.

MEMBER REMPE: -- Phase 5 is occurring, it would be something I'd like to see included in the discussion.

MR. STECKEL: We'll take that under consideration.

MEMBER REMPE: Thank you.

MR. STECKEL: We can move onto the next slide. Beyond-design-basis external event. These were the areas of our review: The external hazards; core cooling; containment; spent fuel pool cooling; ventilation; water and fuel supplies; FLEX equipment and off-site resources; power supply; shutdown ops; protection of equipment; mechanical equipment capability; programmatic controls and procedures; and COL information items.

To summarize in the next slide, we found the applicant's approach consistent with applicable guidance and to be acceptable and subject to the verification of our confirmatory items and our DCD -- and the DCD and technical report modifications specified in the RAI responses.

Questions?

(No audible response.)

MR. STECKEL: Moving onto 19.4, loss of large areas. These are the areas of our review: DCD enhancements addressing 50.54(hh)(2) for firefighting, actions to minimize fuel damage, and actions to minimize radiological release; and review for conformance with NEI 06-12 and the guidelines in Revision 3 including firefighting, measures to mitigate damage to fuel in the spent fuel pool, measures to mitigate damage to the fuel in the reactor vessel, and to minimize radiological release. We had -- I believe we had one RAI involved in this. It was answered. There are no open items.

And the next slide shows a quick discussion on mitigative strategy. Applicant addressed each item in the mitigative strategy table and identified those considered COL items and those provided as design enhancements. And our conclusion is with incorporation of the confirmatory items in the next DCD revision staff will be able to conclude that the LOLA DCD section includes the necessary information to find that design enhancements conform to the guidance in NEI 06-12.

MEMBER STETKAR: Jim, as I mentioned when the applicant was up, when I read Section 19.3 and Section 19.4 of the DCD and read the corresponding sections of the staff's safety evaluation, they read as if they are submittals for an apple and a review of an apple and submittals for a tractor and review of a tractor. I'm trying to emphasize the fact that these are not integrated at all.

Your slides confirm that. You said you've reviewed this against 10 CFR 50.54(hh)(2), which will disappear. It won't be existing anymore if indeed the Commission approves 10 CFR 50.155. It's integrated into the new rulemaking. There is any number of different NEI reports that you can cite that have different numbers and different revisions. There's Interim Staff Guidance.

My basic question is how is the staff going to resolve the intent of the proposed rulemaking and its regulatory guidance and Revision 2? If you want to get into Interim Staff Guidance, JLB-ISG-2012-01, the intent of that for an applicant to integrate their mitigating strategies. How are you going to review that before we finally have an accepted DCD? Because right now it doesn't do it in an integrated manner.

MR. DIAS: Good morning. This is Antonio Dias from NRO. I believe I can answer partial -- part of your question.

It is true there are many, many guidances, and everyone is aware that there's a rule that is about to be issued. We just don't know when. So when we were looking at the application, we just made sure that the application made reference to the current guidance at the time.

So when we received the Fukushima report from KHNP, the one that was at the time was the ISG Rev 1 that references NEI Rev 2. That was issued in January of 2016. There's a new revision to the ISG, which is dated February 17. And that's already addressing NEI, endorsing NEI Rev 4. I just opened -- I went up. There's a 4. NEI is very prolific with their guidances. Okay?

So, yes. So it's very confusing. And I'm going to say the question that you pose also is applicable to even to some of the ones already issued. And JLD is taking the role -- the group, the Special Office JLD is taking the role of how to actually perhaps convert some of the license requirements and how that's going to be brought back into a rule once the rule is issued.

But the problem with KHNP is that we cannot wait and see exactly how the rule is going to be issued and when, so we are basically moving forward with guidance that currently existed. In the case of KHNP, because the latest revision of the ISG came in after the report, we are not going to ask them to go back and address that revision to the NEI, or the revision to the ISG. So it's kind of a little bit of a who-goes-first situation.

But what you say, that there seems to be discrepancies between 19.3 and 19.4, that's a very good point, and I think we should take a look at this.

MEMBER STETKAR: I know you have a dilemma. On the other hand, the applicant ostensibly is one organization and the NRC is ostensibly one organization. And if I look at things in a DCD that have a functional capability and in one section of the DCD where that functional capability is specified as having -- I can't find it in my notes, so I'll make up numbers -- 750 gpm. And in the next section of the DCD the same functional pump is specified as having a capacity of a 1,000 gpm, it strikes me as an anomaly.

And that in the sense of integrating these mitigating strategies somebody ought to identify the fact that that pump in the certified design ought to have a specified capacity. I don't care whether it's 750 or 1,000. The capacity ought to be based on functional requirements and whether that pump is used for an administrative licensing requirement in Section 19.4 or whether it's used for the administrative requirement in Section 19.3. We ought to have clarity on what the pump is so that when the COL applicant goes out to buy a pump, he doesn't have to buy two and he doesn't have to figure out whether it ought to be somewhere halfway between 750 and 1,000 gpm.

MR. DIAS: Right. I understand.

MEMBER STETKAR: So I don't know where -- and that I thought was the intent of the rulemaking, to integrate the mitigating strategies for all of this.

MR. DIAS: You're correct.

MEMBER STETKAR: It's why 50.54(hh)(2) was rolled into the new rule.

MR. DIAS: That's correct.

MEMBER STETKAR: So from that perspective, from that technical functional perspective why isn't the staff asking these questions about -- in 19.4 why does the same pump have a different functional capacity than in 19.3? It seems you could do that without pointing to --

MR. DIAS: This is --

MEMBER STETKAR: -- specific guidance, because the guidance doesn't necessarily tell you how many gpm --

MR. DIAS: This is a little bit -- this is -- the fact is that -- first, I can tell you 19.4 was reviewed by the staff, by a separate staff member --

MEMBER STETKAR: I -- that's why --

MR. DIAS: -- way before 19.3. This is not the correct answer.

MEMBER STETKAR: That's why I said ostensibly.

MR. DIAS: This is not even an excuse. It's the truth.

MEMBER STETKAR: Okay. I hope then --

MR. DIAS: We are going to look into this.

MEMBER STETKAR: -- before Phase 5 and the final version of the DCD and the final version of the SER we come to some sort of convergence here on these two sections, because I think that they can be made consistent, at least from a functional perspective, despite the morass of regulatory guidance that we're in right at the moment.

MR. DIAS: Okay.

MEMBER STETKAR: Okay. Thanks.

MR. DIAS: Thank you.

MEMBER STETKAR: And thanks for being honest about the different reviewers.

MR. STECKEL: We're moving in that direction. We're not quite there yet.

MEMBER STETKAR: I just wanted to get it on the record because it is frustrating given the intent of what the agency and the industry in the United States have been trying to do since Fukushima to try to integrate a lot of these bits and pieces that have evolved over the last almost 20 years now.

MR. STECKEL: Understood.

MR. WARD: Yes, we -- I'm taking notes. We'll review the record after this to make sure we include those in our Phase 5 meeting.

MR. STECKEL: Move onto 19.5, aircraft impact assessment. These are examples of the key design features: Reactor containment; building design; design and location of spent fuel pool; physical separation of Class 1E emergency diesel generators and the alternate AC source; physical separation and design of the emergency core cooling and support systems; pressure-related and non-pressure-related fire barriers.

And we've concluded Section 19.5 meets 10 CFR 50.150(b) due to these reasons: Adequate description of key design features and functional capabilities; adequate description of how the key design features meet the assessment requirements; and the applicant has developed a reasonable formulated assessment and it was performed by qualified individuals.

And as a final note the next page, just to let everyone know that the AIA inspection for APR1400 is scheduled for next month. Any additional questions?

MR. WARD: Okay. That completes our presentation. I did want to bring up a couple of questions that came up earlier, one a question about main control room ventilation, dose considerations. And I know Jim mentioned there was some discussion in Chapter 15. I was just scanning the SER and there appears to be an open item related to some questions in that area. So we'll look into that some more, but it may not be resolved it.

And I think we have somebody else here to help answer the question on pH of seven.

MS. UMANA: That was Mr. Powers' question.

MR. MAKAR: This is Greg Makar from the staff. I didn't hear the question.

(Laughter.)

MR. MAKAR: I'm sorry. I apologize.

MEMBER POWERS: The staff -- the applicant has insisted that his IRWST will remain at a pH greater than 7 throughout the 30-day period following a design-basis accident. And I'm wondering how do we know that?

MR. MAKAR: The reviewer in this case looked at the applicant's calculation of the pH over that 30-day period accounting for the change from acidification primarily from hydrochloric acid and nitric acid generated by radiolysis of air and water and cable insulation and jacketing. There's also a base from cesium hydroxide, but the main contributors to the pH change are the acids.

So looking at the total volume and the acid that is generated, base generated and the buffering capacity of the tech spec amount of trisodium phosphate -- those are the main parts of that calculation.

MEMBER POWERS: The questions are in the details here. I mean, there are a couple of other questions that are not so much detail, but where do I see what G values they used for the radiolysis components on this, how they handled HCL production from the cable insulation and things like that?

MR. MAKAR: It's in -- there is a calculation that was submitted. I think it's part of the RAI response. And I think that identifies the G values, which are taken from a NUREG/CR. I'm sorry I don't remember the number of that report, but that's --

(Simultaneous speaking.)

MEMBER POWERS: In other words, something I have not seen? I guarantee you I haven't seen it.

The other question that comes up is do they account for carboxylic acid formation from any organic contaminants? And we've certainly seen that in the RTF tests done in Canada. We know that happens. We know that by experiment and we know that by theory.

MR. MAKAR: Probably not.

MEMBER POWERS: And do we account for precipitation of phosphates due to contaminants coming in to the IRWST? Any calcium, iron, zinc, something like that.

MR. MAKAR: So a loss of phosphate that's assumed to be present for the buffering?

MEMBER POWERS: Yes.

MR. MAKAR: I don't --

MEMBER POWERS: And if you drop it at a solution then it's not buffering anything.

MR. MAKAR: Probably -- that's probably not counted.

MEMBER POWERS: Well, it would still be good to see the calculations.

MR. WARD: We'll look up those references and provide those to Chris Brown to make sure --

(Simultaneous speaking.)

MEMBER POWERS: That would be terrific if you could. I mean, I don't know that there's anything wrong with it, but Id' sure like to see it. Because for instance, just in the area of G values on Hypalon I know of three measurements. They only differ by a decade. And one of them is -- in fact has a qualitatively different progression of HCL production. Similar differences on nitric acid formation rates.

MR. WARD: Okay. We'll get you those references.

Any other questions?

(No audible response.)

MR. WARD: Well, this concludes our presentation for today. Thank you.

MEMBER BALLINGER: Well, thank you. I think now it's appropriate to take public comments since we're finished. And so, I would ask if there are -- is anybody in the room that would like to make a public comment?

(No audible response.)

MEMBER BALLINGER: And we're trying to get the -- is it open? So are there -- is there -- are there any people out on the line that would like to make a public comment?

(No audible response.)

MEMBER BALLINGER: Hearing none, then we

need to have -- ask if there's any other Committee discussion.

(No audible response.)

MEMBER BALLINGER: That not being the case, I turn it back over to the Chairman.

CHAIRMAN BLEY: Thank you. Before we recess for lunch, you need some work on the letter before we go through it?

MEMBER BALLINGER: We have the letter. We have a draft letter. Based on some discussions that we've had here, it would be a good idea to get another crack at doing it, but we could do that either offline or --

CHAIRMAN BLEY: I think that's probably a good idea, so I don't think we'll read it at this time.

Okay. We will recess until 1:00 when we will take up the next item on the agenda.

(Whereupon, the above-entitled matter went off the record at 11:35 a.m. and resumed at 1:00 p.m.)

CHAIRMAN BLEY: The meeting will come to order. At this time, we're going to take up the issue of Browns Ferry extended power uprate, and I will turn the meeting over to Dr. Rempe.

MEMBER REMPE: Thank you, Mr. Chairman.

So on May 3rd, our Power Uprates Subcommittee reviewed the extended power uprate license amendment request for Browns Ferry Units 1, 2, and 3, and the associated draft -- the NRC staff's draft safety evaluation report. And at the end of the meeting, our subcommittee recommended that this LAR be presented to the Full Committee.

Several aspects of this EPU request are of special interest. The three BWRs are Browns Ferry have relatively high power levels. Each unit is a GE BWR/4 housed within a Mark I containment. Previously, Browns Ferry has been approved to use AREVA ATRIUM 10 and 10XM fuels, and in this request, the licensee, Tennessee Valley Authority, has proposed to eliminate the reliance on containment accident pressure. And prior to implementation of the EPU, TVA will install a new replacement steam dryer in each unit.

Today, we are going to hear from TVA, NRC staff, and possibly some of their consultants. Part of the presentations will be closed in order to discuss information that is proprietary, and before I start by turning this over to Kathryn Brock of the NRR management, we need to have a special statement by our member, Jose March-Leuba.

MEMBER MARCH-LEUBA: Yes. I just wanted

to say that I am conflicted on this topic because I helped the staff work on it before I came into the ACRS, so I will restrain myself from doing any deliberations.

MEMBER REMPE: Thank you. So Kathryn?

MS. BROCK: Thank you. Well, my name is Kathryn Brock. I am the Deputy Director in the -- oops. My name is Kathryn Brock. I am the Deputy Director in the Division of Operating Reactor Licensing in the Office of Nuclear Reactor Regulation.

The objective of today's discussion is to present the ACRS a request by the Tennessee Valley Authority to perform extended power uprates at each of the three units at the Browns Ferry Nuclear Plant. We feel we had a successful subcommittee meeting, and we appreciate the opportunity to brief the Full Committee on this important licensing action.

Attendees today do include NRC staff and contractors, TVA staff, and TVA contractors and vendors, and for the first portion, members of the public. We welcome all of you and thank you for joining us.

Next slide, please. As was mentioned, all three units at the Browns Ferry are General Electric boiling water reactors of BWR/4 design with Mark I containments. Browns Ferry is located on the Tennessee River in northern Alabama. Each unit has received a renewed license with a license expiration date in the 2030s.

The proposed EPU power level of 3952 megawatts thermal represents an increase of approximately 14 percent above the current license thermal power level, and approximately 20 percent above the original license thermal power level. The review of the EPU license amendment request has involved over 25 technical staff from 14 different NRR branches, so it's a big effort.

The NRR document RS-001 provides guidance for performing reviews of EPU applications. TVA followed the format and guidance delineated in RS-001 in the safety analysis reports contained in the EPU application. The NRC staff review also followed this guidance, as well as applicable regulations, regulatory guides, and the Standard Review Plan. I thank the staff and the contractors for their thorough and timely review of this important amendment request.

After this extensive review, there are no open items, so the NRC staff recommends approval of TVA's extended power uprate amendment request. And to give you the details of the staff review, I turn the presentation over to Farideh Saba, the NRC project manager responsible for this review.

MS. SABA: Good afternoon. My name is Farideh Saba. I am the Regulatory Licensing Project Manager for Browns Ferry Nuclear Plant Units 1, 2, and 3 in the Office of Nuclear Reactor Regulation, Division of Operator Reactor Licensing.

TVA submitted the proposed Browns Ferry EPU in September of 2015. The NRC staff accepted the application for detailed technical review in a -- a letter dated January 11th, 2016. The NRC staff has planned for completion of the review in July of 2017, which supports TVA EPU implementation plans during the refueling outages in spring of 2018 for Unit 3, in fall of 2018 for Unit 1, and in the spring of 2019 for Unit 2.

Next one? With respect to the agenda, during the open sessions, TVA will provide an EPU background presentation of the planned modifications associated with the EPU, an overview on elimination of credit for containment accident pressure (CAP), and an overview of the steam dryer replacement RSD. Following TVA's presentation, the NRC staff will present staff's review and conclusions regarding fuel and reactor system analysis, review of elimination of

CAP credit, and review of replacement steam dryer analysis. The staff presentation will be closed to the public due to proprietary nature of the information that will be discussed.

If there are no further questions, I would like to turn it over to Mr. Ed Schrull of --

MEMBER KIRCHNER: I have a --

MS. SABA: -- TVA.

MEMBER KIRCHNER: -- question, if I may?

MS. SABA: Sure, yes.

MEMBER KIRCHNER: I am just noticing that your presentations -- TVA's will be open, but your presentations will be closed. Is there some summary of your evaluation that you can make it an open session that the public can participate in? I believe that -- I think it was Dr. Powers pointed out, I think he used the word, you know, is this EPU prudent in light of Fukushima? So it would seem to me that you should present something about your SER in an open session.

MS. SABA: I understand your point, and actually, when we were planning, the reason we did it like that, because some of the review done by the staff is proprietary. So we planned it that way, that any question that public can hear can be asked during

TVA presentation, and either TVA provide the response or staff will help. Anything also that is asked during TVA presentation that we see that as proprietary, they can refer it back to the time that we are doing the NRC members, they are doing it. This way, we are separating proprietary information from public information, because also, we have very short period of time. It's not like -- it wasn't like subcommittee.

MEMBER REMPE: Before -- again, I think I would like to talk about your response to your question, but let's let the member of the -- is -- did you have a question, too?

MS. SABA: He is Ed Schrull. He is coming after me.

MEMBER REMPE: Oh, okay. So I think you have heard them say there are no open items, and they are recommending that it move forward, and I think that as you go -- I mean, you were there in the subcommittee meeting, and so they -- yes. So I think that that is sufficient, in my judgment, but if there is something that you would like to -- that you believe is open that should be said, why don't you tell us, if there is something that you really wanted to have said?

CHAIRMAN BLEY: Or you could revisit just before we go into closed session.

MS. SABA: Right.

MEMBER REMPE: Okay?

MS. SABA: That is exactly what was planned. Thank you very much for your question. Thank you.

So I -- I would like to turn it over to Mr. Ed Schrull of TVA for an overview of the EPU for Browns Ferry.

MR. SCHRULL: Thank you, Farideh. My name is Ed Schrull. I am the TVA Corporate Fleet Licensing Manager. For major projects such as the Browns Ferry Extended Power Uprate Program, TVA forms a dedicated site team that exists for the duration of the project. TVA corporate then performs an oversight function for the project.

So I would like to introduce the members of the Browns Ferry EPU leadership team that are seated at the front table. Starting from our right to left: on the right side there is Mr. Dan Green, who is a Senior Licensing Manager of the EPU project; next to Dan is Mr. Gerry Doyle, the Director of the EPU project; and next to Gerry is Mr. Pete Donahue, the Senior Engineering Manager of the EPU project. All three of these gentlemen have been with the Browns Ferry EPU project since its inception. Next to Pete is Mr. Lang Hughes, the Browns Ferry General Manager of Site Operations; and next to Lang is Mr. Steve Bono, the Browns Ferry Site Vice President.

At this time, I will turn it over to Mr. Gerry Doyle, the Director of Extended Power Uprate Project.

MR. DOYLE: Thank you, Ed. I appreciate that.

Thank you, and good afternoon, Mr. Chairman, ACRS Committee members, Director Brock, and the NRC staff, good morning -- or, sorry, excuse me, good afternoon. My name is Gerry Doyle, and I am the Director of the EPU project. I wanted to introduce and provide the background of our team and to share the project goals we established early on to work effectively with the NRC staff.

As part of our preparation, we performed extensive benchmarking of previous EPUs, including Grand Gulf, Monticello, Nine Mile Point, and Peach Bottom, including attending both Peach Bottom ACRS committee meetings. Our EPU project team is comprised of personnel having extensive boiling water reactor plant experience and specific EPU experience.

Our team includes dedicated project and plant TVA staff; major vendors, including GE-Hitachi and AREVA; as well as experienced specialty contractors. We were fortunate in that we were able to take full advantage of a wealth of industry operating experience and experienced, qualified, and talented experts. We feel we have assembled the industry's best and expect to demonstrate that today. Next slide, please.

One of the first things we did was align our project on high-level team goals, recognizing the need to work closely with the NRC -- with our NRC counterparts and demonstrate we were committed to implementing an industry-best EPU, specifically, to present the NRC, Nuclear Regulatory Commission, with a high-quality license amendment request consistent with RS-001 standards. To that end, we have worked closely with Farideh and the technical reviewers to ensure we were providing the necessary information that they needed for their review in a timely manner: to resolve containment accident pressurization issues, which Pete Donahue will discuss further along in the presentation; to install replacement steam dryers for all three units, again, a separate discussion by Dan Pappone of GE as part of this presentation; and finally, I wanted to ensure we provided a smooth transition to EPU for plant operations, such as there were no surprises or issues in running or maintaining the plant. Those are our high-level team goals.

With that introduction, I would like to turn it over to our Site Vice President, Mr. Steve Bono.

MR. BONO: Good afternoon. I want to thank the Committee for the opportunity to present our EPU project today.

Much of my slide has been presented, so in fairness to Committee time, I will just give a quick summary. But all three units are BWR/4 GE models with a Mark I containment. You can see our original operating license dates, and you can see where we renewed that license. So all three units today currently operate under a period of extended operation. We did implement a stretch uprate of 105 percent original licensed thermal power, and what we are requesting today is a proposed 120 percent of the original licensed thermal power limit.

And with that, I am going to turn it over to our General Manager of Site Operations, who is going to go through some of the design changes and modifications that we implemented to implement the EPU project.

MR. HUGHES: Okay. Good afternoon. My name is Lang Hughes. I am the General Manager of Site Operations at Browns Ferry. I will go through some of the modifications, many of which have already been completed across all three units to support the extended power uprate.

As you can see on the slide that is currently on the screen, many of the modifications that we have with these circles that are colored in have already been completed. Included in this are the replacement of the original design pumps for the condensate booster and the reactor feed pumps. We have the -- the EPU pumps installed. The Unit 1 and 2 pumps have all been installed in operation since 2007, but we completed installation of the condensate booster pumps and the reactor feed pumps on Unit 3 in the spring outage of 2016.

Many of these major modifications our operators have been trained on for some time. You know, it is nothing new with regard to individual component operation for them. This is something that, you know, we feel strongly about is the operator training piece, and having these pumps in service for some time has been able to let us run through EPU scenarios on the simulator with this installed equipment and so forth to get our team ready for full implementation.

Next slide, please. I won't go through and talk about all the modifications. A couple of them that I will mention, we will talk later about the replacement stream dryer. As far as the mods, you can see them listed on the slide there. Really, these are all -- for the most part will be transparent to the operators. Main generator hydrogen pressure, we will operate at a higher pressure. One thing that we are doing with this modification in addition to assist the operators is automatic hydrogen pressure control, and remove the need to manually make up the hydrogen pressure in a normal band. It will be controlled for them, and they will have the ability to control manually if there is an issue with the automatic control system.

Next slide. Other modifications, the hardened wetwell vent modification, we have completed the installation on both Units 1 and 2. Unit 3 will be installed in the spring outage of 2018. The other one that I will address will be the alternate leakage treatment pathway. That is -- you know, for us, it is something that, you know, we are at a non-conforming condition with respect to the alternate leakage treatment pathway, and we do have designs completed. They will be installed in the upcoming EPU outages. Unit 3 will be the first in the spring. These will install air-operated valves that will fail open, as required.

We're all installing a new valve in the main steamline drain system as well as replacing four motor-operated valves with air-operated valves, also in the steamline drain system. So this will remove the non-conforming condition prior to EPU implementation, and also, for this, this is currently an operator workaround that we have in place for the steamline drain system, which we'll move that as well in the Unit 1 and Unit 2 outages to follow. Next slide.

For the next slide, a couple things I will say with respect to the instrumentation and for the control room operators. We will be, you know, making some changes to that associated with set points and respanning the instrumentation. The control room, for the operators, what they see, it will be transparent to them. They are going through now and looking at all of the new EPU conditions on the simulator as far as, you know, what these changes will be.

The one thing that we are going to do as far as modifications to really assist our operating crews is we're going to install a new condenser vacuum instrumentation. Right now, you know, we have condenser vacuum indication, but we're going to go for each individual condenser to where the operators will have, you know, specific points to monitor for each individual condenser. Air condensers are designed a little bit different than some plants. They have a common area, but they also have three separate water boxes, either one of which can provide a turbine trip signal, so this installed upgraded instrumentation will give them exact indication on each individual water box. This is something that the operators have asked for, that it's important to them, so we will be doing these implementation modifications in the Unit 3 outage and each succeeding EPU outage to follow.

So pending any questions, this completes my overview for the modification summary.

MEMBER SKILLMAN: Yes. I would ask this: with the condenser pressure model, with the higher flow rates, are you going to be pushing those condensers to where they are in fact a trigger for your higher power conditions?

MR. HUGHES: With higher flow rates, there will be some, you know, change with respect to condenser vacuums, specifically in the summertime, when we have elevated cooling water temperatures. Now, the one thing about it as far as, you know, our procedural requirements for, you know, turbine loading and power for condenser vacuum, those will not change. You know, we still have trigger values in our procedures and so forth.

The other thing that we have done, not really associated with EPU, but for margin purposes, is, you know, over the last couple years, we have started the condenser cleaning -- or chemical cleaning of condensers, which we, you know, did some industry benchmarking and started that, which has really improved our margin with respect to condenser vacuum. What this does is give the operators a better indication for each water box, which actually assists them for their trigger values in accordance with our procedures. But as far as, you know, operating margins, those will not change.

MR. HUGHES: Okay. The reason for my question was are you actually pushing yourself into a condition where you're more likely to get trips, particularly when the -- when the Tennessee River is running at its warmest?

MR. DONAHUE: We performed analysis, both pre-EPU and -- and EPU. Another modification that we will be performing is that we're -- we're changing the alarm set point, and also the -- the trip set point through GE analysis for those -- those alarm and trip set points, and when you look at both of them together, we will actually be at a similar position that we are today. So the EPU does affect condenser vacuum. However, the alarm and trip set points offset that.

MEMBER SKILLMAN: Okay. Thank you.

MR. DONAHUE: So I am Pete Donahue. I am the Senior Engineering Manager for EPU. I will be presenting elimination of -- of credit for containment accident pressure.

So I am on Slide 15. Slide 15 shows a general diagram of the -- the terms that go into determining the available net positive suction head so that we can compare them to the required net positive suction head to ensure that we -- we are not taking any credit for containment accident pressure. Next slide.

MEMBER REMPE: Pete?

MR. DONAHUE: Yes.

MEMBER REMPE: I know you said it during

our subcommittee meeting, but just to make sure that every member here has it and that I have it correct, you indicated during the subcommittee meeting that the current analysis relies on the maximum CAP credit of 3 psg for about 80 minutes during LOCAs, is that a true statement?

MR. DONAHUE: That is, it is 80 minutes for the long-term, and about 10 minutes for the short-term, so a total of like 88 to --

MEMBER REMPE: Okay.

MR. DONAHUE: -- to 90 minutes.

MEMBER REMPE: Thanks.

MR. DONAHUE: So Slide 16 goes over the -- some background for containment accident pressure. What we did is we took -- we took the opportunity to eliminate crediting containment pressure during -- during an accident or an event, and there's really two types of events, right? There's special events and there's design basis accident types of events. So -- so we -- we performed our analysis to -- to ensure that we're not crediting containment accident pressure. As a matter of fact, we're going to update both our design and our licensing basis to show that we do not credit containment accident pressure, and that statement will also show up in our technical specifications.

So -- so this -- we -- we used the -- the current guidance, that is, SECY-11-0014, to perform this analysis. Next slide. The -- the -- this slide shows -- this Slide 17 shows some of the -- the aspects of -- of the analysis that we performed in order to eliminate crediting containment accident pressure and some actions that we have taken. The -- the first shows that we did perform a number of comprehensive containment analyses using a program that some of you may be familiar with, SUPERHEX, analysis to perform containment accident pressure.

SUPERHEX really just gives us the temperature in which we will evaluate the vapor pressure term for the net positive suction head, so we have done a number of analyses, you can imagine, with different initial conditions and different events, so there is -- there is -- and with different flow rates.

We also increased the -- what is called the -- the heat -- the heat exchanger K-value, and if you're not familiar with this heat exchanger K-value, it is -- it is really a factor that you use in doing the -- the containment -- heat transfer analysis, where you can apply this factor to the difference between the hot side and the cold side, inputs to the heat exchanger. So this K-value is no more than that, but -- but there's a pretty complicated formula to determine that --

CHAIRMAN BLEY: So is this analysis or physical change?

MR. DONAHUE: This is analysis.

CHAIRMAN BLEY: Analysis? Okay.

MR. DONAHUE: The -- so -- so in order to ensure that we maintain that -- that K-value, we are putting that information on performance of our heat exchangers into technical specifications to ensure that we -- we both perform the -- the heat exchanger performance testing on -- on a given frequency and that we maintain the -- the K-factor that is in our design basis.

MEMBER REMPE: So Pete, just to clarify a couple of things, first of all, this response to Dennis's question, it is analysis, but isn't it based on fouling factors that were measured? So before you justified this K-value, you actually had some data to support it?

MR. DONAHUE: Oh yes, yes.

MEMBER REMPE: Okay.

MR. DONAHUE: We validated the -- the fouling factors with performance tests on all 12 of our heat exchangers.

MEMBER REMPE: And then to respond to Member Kirchner's question, I don't think it is proprietary to say that this technical specification was a result of interactions with the staff and that they -- because they want to make sure they have confidence in the analysis, that they have imposed this --

MR. DONAHUE: Absolutely.

MEMBER REMPE: -- upon the licensee.

MR. DONAHUE: Right.

The next thing we did was we increased the boron-10 enrichment during the anticipated transient without scram special event to ensure that we had adequate margin for containment accident pressure in that event.

MEMBER SKILLMAN: Pete, you didn't include at this bullet that you increased the flow rate for the --

MR. DONAHUE: We also, in the technical specifications, there is a specification for where flow rate is an element of determining the acceptability of -- of the ATWS and the injection rate. I am not sure we had a bullet for -- for it.

MEMBER SKILLMAN: I am not sure it was

communicated, but you did in fact increase the flow rate to --

MR. DONAHUE: Right.

MEMBER SKILLMAN: -- achieve what 774 --

MR. DONAHUE: Right.

MEMBER SKILLMAN: -- ppm in the time requirement for the 1 percent shutdown.

MR. DONAHUE: Right.

MEMBER SKILLMAN: Okay. Thank you.

MR. DONAHUE: And we also used the -- the vendor supply net positive suction head curves where our design basis didn't have that information previously before we performed this analysis in changing our design basis, and we utilized -- for initial conditions, we utilized all technical specification initial conditions with the exception of the fire special event, where we used nominal values in accordance with the SECY guides.

MEMBER REMPE: Pete, again, just to make sure that we have it clear, when Member Skillman said you increased the flow rate, he means in the analysis. My understanding is that you used a more realistic evaluation of what the flow rate would be. You didn't actually increase the flow rate, you increased it in the analysis --

MR. DONAHUE: Oh yes, absolutely --

MEMBER REMPE: -- okay?

MR. DONAHUE: -- the pumps didn't change.

MEMBER REMPE: Is that understood? Okay. Thank you.

MR. DONAHUE: Okay.

And then finally, on Slide 18, the analysis did demonstrate that we did not need to credit any containment pressure as a result of an accident or the event. That's it, I think. Any questions?

(No audible response.)

MR. DONAHUE: All right. I will turn it over to Dan to talk about the replacement steam dryer.

MR. PAPPONE: I am Dan Pappone. I am GE-Hitachi, and I have been -- for lack of a better word, I have been kind of the principal investigator for the steam dryer issues since the first indications of failures at Quad Cities following their power uprate in 2002.

Next slide, please. So it has been a long trip, but the replacement steam dryers we're putting in in Browns Ferry is really the end product of all of that research and development that we have done. They are based on the curved hood prototype replacement dryer that we put into a BWR/4 reactor for their extended power uprate about ten years ago, now, so we have had a lot of operating experience with the design.

The design itself, what we found coming out of the Quad Cities was that these fatigue stresses, the flow-induced vibration, were really tight on that with the original dryer designs, and we basically had to refine our steam dryer design because of that. Because of that, the -- these replacement dryer designs are much -- much more robust than the original dryers. The original dryers, the fatigue -- fatigue stresses were believed to be negligible, and we did have some cracking initially, and that was patched. It went on, but when we started doing the extended power uprates, we found -- we found we were right up against the structural limits, so that is basically what has driven the industry to replacement or -- or extensive modifications.

Key to that is -- key to being able to do those design analyses is developing the acoustic load pressure methodology for putting on the structure in order to do those predictive structural analyses to be able to do a design efficiently, and the method that we have come up with, we take measurements -- for a plant going in, we will take measurements from the main steam lines, basically using the main steam lines like stethoscopes to listen in and develop -- capture the frequencies that are actually loading the dryer, and then using that to develop the acoustic load definition that we then put on a structural model.

And methodology that we developed, we used that full methodology end-to-end on the previous replacement dryer project. The results of that were good, and we have also captured that process as part of the ESBWR design certification documentation. So we have captured that methodology. It has been reviewed and accepted by the NRC on those projects.

And then of course, we have always analyzed the dryers for the primary structural loads to make sure that the dryers maintain their structural integrity, don't generate any of those parts during normal operation transient and then accident events. Next slide.

MR. RICCARDELLA: Which plant was the prototype BWR for?

MR. PAPPONE: Those were the Susquehanna dryers.

MR. RICCARDELLA: Thank you.

MR. PAPPONE: So in the replacement dryer world, we have replaced the stream dryers for Quad Cities Units 1 and 2 and then used that same design on the Dresden's Units 2 and 3. The next -- next EPU application was really modifications that we did to Vermont Yankee where we put the repair modifications that we had developed on the original Quad Cities and Dresden dryers while we were going through the replacement dryer design process, we had put that set of repairs on Vermont Yankee. Those repairs performed successfully for the EPU life of the plant with no issues.

We went back to the curved hood dryer -- basic curved hood dryer design for the Susquehanna replacement dryers partly because of fabrication issues. Complexity of the designs we used for Quad Cities and Dresden, we had a lot of castings and the like. We wanted to move away from that and get back to simpler fabrication techniques.

And then Grand Gulf was the next step, again with the curved hood dryer. So in each of these steps, we are taking whatever lessons learned, folding those into the next design. And Browns Ferry is -- is taking advantage of all of this experience in their replacement dryer designs. Next slide.

So we did have some -- some changes to the

-- from prototype design when we went to Browns Ferry. We did have -- because of the hold down differences, the dryer hold downs, in Unit 1 versus Units 2 and 3, we did have to accommodate those differences and come up with a slightly different mechanism for holding the dryer down in the event of a steamline break accident that would push the dryer up, and we wanted a single design that would accommodate all three units.

We did fold in the operating experience and lessons learned from the previous analyses, both in the operating world and in the fabrication world. And we went through a couple of design iterations specifically for Browns Ferry. First is when we put the specific load definition on, we see what parts of the dryer are responding to the frequencies that we're experiencing, take care of any amplified structural response we may see, do some design changes to -- to address those, and then with the Browns Ferry project schedule, we had the time to do a full second design iteration where we were able to go through the whole structure, even out the load paths, even out the stresses, and bring in as much margin as we could into the design.

And next slide. This flow chart gives an overview of the analysis approach. We've got two finite element models: one for the acoustics where we have modeled the steam dome cavity with the dryer in -- in the steam dome, and that is the tool that we use for developing the acoustic load definition and projecting the loading onto the dryer surfaces. We've got a second finite element, ANYSYS model of the dryer structure itself, basically a shell model, and that is the primary analysis for generating the -- generating the stresses, calculating the stresses.

The one we do take going in, because we don't have on-dryer measurements of the acoustic loading for Browns Ferry, we are starting with the steamline measurements as a basis for our load definition. We -- the one key load component that is not present at the current power is if there is an -- if there is an acoustic resonance in the safety relief valve standpipe that may come as we raise -- increase power, we don't have -- we may not have the measurement of that today. We have to go through, look at the characteristics, the geometry of the steamline, what we're seeing in the actual acoustic signature in the steamline measurements we have taken, and predict whether or not we will -- we have the possibility of the resonance coming up, and if so, what frequency, what amplitude.

We go through that process and factor that into our acoustic load definition that we're putting on the structure. Take -- come up with the -- we also go and project the amplitude growth for the loading when we get up to EPU, take all of that information for the load definition, put that on the structural model, run that -- run that analysis, get the raw stresses, then go through several stress adjustment factors, scaling the stresses up to EPU. We have bias and uncertainties that we apply to the raw results coming from the benchmarking where we benchmark methodology against instrumented dryer structural response, so we factor that -- that in.

There are operating conditions that may increase the stresses from non-acoustic sources: vane passing frequency from the recirculation pumps can drive structural vibrations that carry into the dryer and increase the stresses by a small but non-negligible amount. And then that -- that will give us our final flow-induced vibration fatigue stresses. We compare that against the ASME code fatigue allowance limit. We are looking at the far end of the fatigue curve because we are basically designing assuming we've got those loads acting for the whole time -- for the remaining life of the plant, and that then determines how much margin we have to the fatigue limit.

Those fatigue analysis results also get carried into the acceptance criteria for the -- for the limit curves for power ascension, and the analysis itself is part of the framework that we use for monitoring -- predicting and monitoring the dryer stresses as we're coming up on power ascension. The -- those results are also fed over into the primary stress analysis for the normal upset faulted condition primary structural analyses, and then we compare those results against the ASME code acceptance criteria for those conditions.

Next slide. So when we got through the fatigue analysis, we were basically at a factor of two margin to the code limit for the -- for the fatigue limit. We've got one component that was just a hair under two, and so that was successful. That design is successful. It's nice going into the power ascension with this kind of margin. Yes?

MR. RICCARDELLA: So as I understand it, this factor of two applies to the initial analysis up to EPU conditions prior to the benchmarking, and then once you do the -- and then benchmarking, you're allowed to reduce that to one. Just out of curiosity, on the Susquehanna plant, how much of that difference between two and one was used up with the bias, the new bias in uncertainty factors?

MR. PAPPONE: By the time we got -- got all the way through -- through the -- through the process, we were still -- we were still above the limit. There was also some history in there in that when we discovered errors and had gone back and checked to confirm the operability for the previous analyses, that -- but we're still above that -- above the ASME limit.

MR. RICCARDELLA: Yes, but it was at 1.01, 13 or 1.1?

MR. PAPPONE: I don't remember --

MR. RICCARDELLA: Ballpark?

MR. PAPPONE: I don't remember -- well, it was about 1.3 when we were done with the power ascension.

MR. RICCARDELLA: Okay, okay.

MR. PAPPONE: But that -- in that case, we were going in almost with -- with a blind set of analysis tools because that was really the first time we had gone through that process.

MR. RICCARDELLA: Yes. So we should do better on --

MR. PAPPONE: We should be doing -- we did --

MR. RICCARDELLA: -- on this one.

MR. PAPPONE: We did quite a bit better on Grand Gulf, so --

MR. RICCARDELLA: And for the record, that ASME code limit has a factor in it, a factor of two in it also, right?

MR. PAPPONE: Yes.

MR. RICCARDELLA: Thank you.

MR. PAPPONE: Actually, it's all of this margin that we're putting in at the beginning that is driving the replacement dryer, because we do have dryer -- we do have plants that were -- had gone up to EPU before with the original dryers with no modifications, and there haven't been any issues there. But it is the analysis, predictive analysis world in the margin that is -- that is driving us to this kind of a robust design, and the fact that we don't want to have another Quad Cities.

So next slide, please. Now for the primary stress evaluations, these are straightforward. The original dryers were designed for the faulted condition with the intent that if we had that margin available to accommodate the accident loads, we had plenty of margin for the fatigue. Well, things have kind of flipped a little. Now that we have gone and put in all of this additional margin for the fatigue, we've got plenty of margin for the accident conditions.

And we'll go through the process. And dryer is not a safety-related item. It has no active safety function. We just need to make sure that during an event, it maintains its structural integrity, it doesn't generate any of those parts. And then it's not a code component, it's not of course a core component, but we're -- for all intents and purposes, we are designing it as if it were a code component.

Next slide. The power ascension monitoring process we have also worked out over the replacement dryer projects, and the process is similar. For the lead unit, both the replacement dryer and the steamlines will be instrumented, and as we come up in power, we will be able to do the benchmarking, the Browns Ferry-specific benchmarking, of the analysis, come up with Browns Ferry-specific bias and uncertainty factors that will then fold into the power -- the limit curve monitoring and also into the follow-on units.

The power ascension limit, the power ascension limits for lead unit will be on the dryer instrumentation, and when we do it, the main steamline will be backed up in case we -- in the event that if we lose too many instruments on the dryer, we will be able to continue. And that brings -- that gets us to the point of the factor of two that you had mentioned. If we get through that Browns Ferry-specific benchmarking, at that point, we will be working with a benchmark model at that point, the power ascension.

CHAIRMAN BLEY: The Browns Ferry units, I don't remember. I thought they were built at somewhat different times.

Is the piping identical as you go from unit to unit?

MR. PAPPONE: It's virtually identical. There is one safety ruling valve standpipe that's in a little bit different position.

But they're virtually identical as far as the loads. And that shows up -- going into the design, we had measures in all three units. And they are quite similar.

CHAIRMAN BLEY: Thank you.

MR. PAPPONE: When we get up to -- when we get up to the fully computed power in the lead unit, we'll be performing -- taking data and performing benchmarking there.

And performing a full scope, confirmatory structural analysis for the FIB loads. And that analysis will then be the basis -- used as the basis for the follow on units for the limit curve development.

The follow on units will only have the main steam lines instrumented. It will be matching -- essentially matching the signatures going up on that and addressing and using the differences in those signatures to account for any response differences we may be seeing.

And then when you get up to the fully fueled for on the follow on units, we'll go through a structural evaluation to confirm that each of those dryers is also meeting the limits.

The extent of that, the follow on evaluation will depend on how close the signatures are to the lead unit.

CHAIRMAN BLEY: I'm curious. You gave us a little history of how the designs evolved from plant to plant.

When you did this kind of testing on the other plants going up, did they all track pretty well from the first unit to the others?

MR. PAPPONE: It did. In fact PPL, the two unit case we had before, where we were going through in this level of detail was for the Susquehanna. The two units in Susquehanna.

And we went through and did the power ascension for the lead unit. PPL basically did the power ascension for the follow on unit.

CHAIRMAN BLEY: Um-hum.

MR. PAPPONE: So -- and they did track quite well.

CHAIRMAN BLEY: Okay.

MEMBER SKILLMAN: Dan, let me ask this. The information on this stream dryer appears very convincing. And I just wonder, can this create a false sense of security?

And are you asking yourselves, what can go wrong?

MR. PAPPONE: Well, I don't think it's a false sense of security. Because we have put in so many over -- so many replacement dryers over the time period.

And we just -- we have not been having any issues showing up. The -- and also go back to the operating history for the original steam dryers, really it's been Quad Cities -- the Quad Cities in Dresden that were in a way an outlier --

MEMBER SKILLMAN: They were the outliers.

MR. PAPPONE: Because of the specific conditions for those plants. And partly, you know, the -- partly the dryer design would push -- that particular dryer design did have a limit.

Those dryers work fine for 30 years until we raised the power. And the weak points showed up.

MEMBER RICCARDELLA: Also as I understand it, there's plans for inspections on the first couple of refueling outages to confirm that they're having problems or not.

MR. DONAHUE: Also during start up, we do have level one and level two criteria. Where we look forward, project the readings that we do get at one power level to the readings that we hope to get at the next power level.

So, that will help us also to make sure that we're not going to get into a power level that we can't -- where we exceed the criteria.

MEMBER SKILLMAN: Thank you. Thanks.

MR. PAPPONE: All right. So the next slide. The next slide is well, during the power ascension we do have hold points for NRC review.

When we get to today's current power, we'll be base lining the methodology on Browns Ferry specific bias and uncertainties. And there's a ten day hold period for NRC in review of that benchmarking.

And then a five percent power increments. Going up we'll have four day hold points. This is the same power ascension review plan that's been licensed conditions for most of the replacement dryer projects.

Follow on units will follow the same basic process. It will be a smaller scope because we won't have -- won't be doing the same benchmarking.

It means we won't have the on dryer instrumentation. Then once each of the units have -- reaches the fully queued power, they'll be a full scope inspection for the following two cycles.

And that's also the same approach we've been using in the other projects. After that long term, the inspection plan lines up with the industry guidance.

Then with that, any other questions?

MEMBER REMPE: But this part is the end of the open part of the meeting. And I think we need to ask if there's any members of the public at this point that want to make a comment?

Or if there's any in the room as well as on the line. And I believe the lines are open.

(No response)

MEMBER REMPE: I see no one in the public or in the audience here who wants to make a comment. And we don't -- I believe we confirm the line is open?

MEMBER STETKAR: Just ask.

MEMBER REMPE: Is the line open? And are there any comments?

(No response)

MEMBER REMPE: While we're waiting to do that, it might be good -- I believe we're going to switch out, right? And so you could go ahead and do the switch out if you wanted to.

(Off record comments)

MEMBER REMPE: Well, the public may -- should I close it before I let the public have their chance to have comments? I think we need to let them confirm there's no public --

(Off record comments)

MEMBER STETKAR: Mr. Brown, it sounds like things are coming online rather than going off. Is the line open?

MR. BROWN: No. The public is closed and closed the lines open now.

MEMBER REMPE: Okay. So were there any members of the public out there when I was calling for it? Was it open?

MR. BROWN: It was open.

MEMBER REMPE: It was open. Okay. So at this point, we are going to close the open session. And I don't have a gavel. Okay.

(Whereupon, the above entitled matter went off the record at 1:51 p.m. and resumed at 3:16 p.m.)

CHAIRMAN BLEY: We are back in open session with a new topic. Westinghouse shutdown seal. So I turn the meeting over to Mr. Stetkar.

MEMBER STETKAR: Thank you Mr. Chairman. I will first announce that we are going to run this session similarly to the one that we just had.

We will have a rather short open session where the staff will provide an overview of their review. And Westinghouse or the PWR Owners Group will provide a brief introduction.

And then we will go into closed session because we need to hear details of the seal design and the topical report and so forth that has proprietary information in it.

Give you, the members who didn't attend

the subcommittee meeting a little bit of the background. The NRC staff back in, I don't recall, earlier, has approved the use of the so called Generation III shutdown seal for use in applications of the flex mitigation strategies.

Saying that, the staff did not approve any models for numerical data for failures of the seal for use in any risk analysis that might support those evaluations that are done in response to the orders.

PWR Owners Group has submitted a topical report that contains information about the logic structure, the PRA models, the success criteria, consequences from success or failure of the seals, and numerical values to be used for failure rates for the seals.

That is what we're talking about today. That topical report on the PRA models and failure rates for the seals. And that's what the SER has written against.

So it's not necessarily the design of the seal itself. Although we obviously have to understand how it works. But, I just wanted to make that clear for the record.

As I said, we had a subcommittee meeting on February 6 where we had pretty extensive discussions about the topical report, the seals.

Since the subcommittee meeting, the Owners Group has submitted a supplement to the topical report. Which I understand will be included in the approved version of the topical report after the safety evaluation is issued.

So that what we're actually reviewing today is the Rev 1 of the topical report that we had available for the subcommittee meeting. Which document has not changed, plus the supplemental information that we've received about a month or so ago now.

And with that, I have a couple of other things to say once we go into closed session. But let's proceed with the open session. And ask Sunil Weerakkody from the staff to start. Sanil?

MR. WEERAKKODY: Yes. Thank you John. Sanil Werrakkody. I'm the Acting Deputy Director in the Division of Risk Assessment. And I really look forward to this opportunity.

As John mentioned, you know, the licensees, a number of licensees have already installed the Gen III, what we call the Gen III shutdown seals. And we are of the opinion that when these seals operate as designed and as expected, the core damage frequencies of these plants reduce. And in some cases, significantly.

Because of that, we are very supportive of efforts on the part of the licensees to implement this modification at their plants. So, when NRC completes the approval of this topical report, the licensees will have and what I would call an acceptable method for crediting these seals when they do their PRAs and certainly their results to us in their risk-informed applications.

So during the last three years, and that's how long we have been developing this topical report, the staff from the Division of Risk Assessment and the Division of Engineering have expended a significant amount of resources to provide a very thorough review of this report. Which included multiple rounds of REIs, numerous meetings, visits to Westinghouse test specialty, and observing post-operational testing.

Then as John mentioned, this year, on February 6, we briefed the ACRS subcommittee. We provided a draft SER. And Westinghouse provided significant details of the design and some of the old designs.

And we received very valuable feedback from the subcommittee. Even though the failure probability numbers did not change significantly, I could say that based on the feedback from the subcommittee and the number of meetings we had with Westinghouse, I believe we have a draft SER. Which is of, you know, higher quality. Which is more comprehensive.

And as John mentioned, also on the part of Westinghouse, they sent a supplemental -- they sent supplemental information which became the basis for us to revise our safety evaluation.

So we really want to thank you, thank the subcommittee for their feedback. And we really look forward doing this particular case. I'm not sure whether the process requires that or not.

But, speaking for the staff, we really look forward to a letter from the full committee after this meeting. That's all I have to say. Thank you and John, I think we go to the staff now.

MEMBER STETKAR: Yes. Please. And by the way, we -- subject to approval by the full committee, we are planning to write a letter on this topical report and the SER.

MS. LYONS: Good afternoon. My name is Sara Lyons. I'm a Reliability and Risk Analyst in the Office of Nuclear Reactor Regulations, Division of

Risk Assessment.

And I'm here with two of my colleagues, Ching Ng to my left is also with the Division of Risk Assessment. And Bob Wolfgang is with the Division of Engineering.

The three of us have been the primary reviewers on this topical report. And so I'll discuss the publically available information briefly here.

But before I get into that, I just wanted to indicate, you know, as Sanil mentioned, this topical report did come to us on July 2014. And we've had a number of, you know, regulatory and technical issues that we've had to work through during that time.

So I'd like to acknowledge some of the other areas where we had insight from the staff. Including our division of Policy and Rulemaking, where they've been helping with project management type of support.

And other offices, Office of the General Counsel, the Office of New Reactors, and the Office of Nuclear Regulatory Research, where they provided some insight for technical consistency. As well as the Reliability and PRA Subcommittee.

And I'd like to echo Sanil's feedback that we did get some valuable insight there. It made changes to our SER as a result of that input. And even added an extra limitation to our SER and are able to bring a higher quality product to the full committee today.

So on slide two, the topics that we're going to discuss are the purpose of the review, the background, the scope of the review, and the PRA model review as well as staff conclusions.

So on slide three, the purpose of this review again, as was stated earlier, it was to review the probabilistic risk assessment model. Which will allow licensees to credit their shutdown seal in risk-informed activities.

The installation of the shutdown seal is an important safety improvement that can mitigate reactor coolant system inventory losses through the reactor coolant pump seals. And a need for reactor coolant system makeup to achieve a stable state.

So on slide four, here's a little bit more background. A loss of seal coolant -- I'm sorry. A seal loss of coolant accident or seal LOCA can occur when the thermal barrier heat exchanger cooling and the seal injection both fail.

According to the currently approved model, the leakage could range from 21 to 480 gallons per minute per pump. And when that shutdown seal is installed and operates as it is intended to, the leakage would reduce to less than one gallon per minute for up to 168 hours.

Which can provide a significant reduction in core damage frequency for some plants. It's really plant specific. But it can provide, as was indicated in an industry study referenced here, up to 50 percent reduction in core damage frequency.

So on slide five, the PRA -- the scope of the review included the PRA model. The topical report PWR Owners OG-14001, Revision One, was submitted for review and approval.

Which described the PRA model. But also referenced a lot of the qualifications, testing information that was provided in a separate document previously, TR-FSE-14-1, Revision One.

The testing information was provided to the NRC staff in response to the Order EA-12049 for mitigating strategies. And the staff had previously endorsed the use of the shutdown seal for the extended loss of AC power evaluations that were performed in accordance with that Order.

This topical report wasn't requesting

approval of installation. That's going to be through other -- performed through other processes such as 10 CFR 50.59.

And the plant specific considerations will need -- there are plant specific considerations that will be addressed in future license amendment requests that weren't -- it wasn't appropriate to cover here because it was very topical, a generic evaluation.

MEMBER SKILLMAN: Sara, on your third bullet, are you communicating that 50.59 may not be used? Or that the topical report is not approving installation and that if a licensee wants to, the licensee may use 50.59?

MS. LYONS: Right. The second one.

MEMBER SKILLMAN: The latter?

MS. LYONS: That latter one. And I think, you know, a lot of plants have already installed these.

MEMBER SKILLMAN: Under 50.59?

MS. LYONS: Under 50.59.

MEMBER SKILLMAN: Okay. Thank you.

MS. LYONS: Um-hum. So, on slide six, the major PRA model review areas included consideration of the Generation I and II shutdown seals, adequacy of the qualification testing, statistical analysis of shutdown seal failure modes, common cause failure, human reliability analysis, inadvertent actuation of the shutdown seal, and performance monitoring.

And we'll go into more detail on those in the closed session. And our conclusion, based on our evaluation of PWR Owners Group 14001, Revision One, as supplemented and various supplements referenced in the AIC.

And used in accordance with the limitations and conditions, we've concluded that the safety eval -- I'm sorry. That the document is acceptable.

And that licensees always have an option of choosing to model their shutdown seal in a different manner. But they would need to justify any differences or deviations from what is approved and their specific risk-informed license amendment request.

MEMBER SKILLMAN: So let me go back to slide five just so it's clear in my mind. You've communicated that a licensee can implement this change through 50.59.

But in the next bullet, you say that plant specific considerations are in a license amendment request. Well, what 50.59 does is it makes the licensee determine whether or not a licensed amendment request is needed or not.

MS. LYONS: Okay. So what I meant by the last bullet wasn't really related to the installation. I can see how that's a little misleading the way we have it on the slide there.

But, the last bullet was really related to the implementation in the PRA. There is some considerations on how the shutdown seal failure is modeled in their PRA.

But it really needs to be handled on a site specific basis.

MEMBER SKILLMAN: So a hardware mode can be evaluated through 50.59 and successfully implemented as a modification. But the PRA implications come out through the LAR.

MS. LYONS: Right.

MEMBER SKILLMAN: Understand. Thank you. Thank you. All right.

MS. LYONS: And that's all we have today. So if there's any questions.

MEMBER STETKAR: Any questions for the staff that pertain to something that we can discuss in the open session?

(No response)

MEMBER STETKAR: Okay. If not, we'll have the Owners Group come up and give us an introduction to the topic.

MR. LINTHICUM: I'm all set. Good afternoon. Once again, similar to the staff, we appreciate the opportunity to come back and discuss our topical report with you.

And explain how we're planning on modeling the shutdown seals in the PRA model. My name is Roy Linthicum. I'm Chairman of the Risk Management Committee within the PWR Owners Group.

We've actually been working on this topical report longer then the staff has been reviewing it. And we are very anxious to get an SE.

A lot of our members really are relying on the SE to be able to move forward with some very important risk informed licensing requests.

Next to me is Matt Degonish. He is -- is that one? There we go. Matt is our primary PRA lead for development of the model.

When we met here on February 6, Matt was unable to attend. He's a new father. So we dragged him away from his family to come down here today.

And Mike LaPresti is here. And he's one of the principal design engineers responsible for the shield.

My presentation will be even shorter than the staff's until we get into the closed session. So, just a quick overview.

The shutdown seal was developed by Westinghouse to limit the leak rate from the RCPs following a loss of all seal cooling. It allows us to respond to a wide range of events such as station blackout and things that would disrupt the power supplies and cause a loss of all cooling.

The important thing in my mind is not only do we get a significant benefit in our PRA models, where we're talking orders of magnitude benefit in the probabilities of seal leakage, but more importantly, we don't challenge the operators as much when we have the seal in place.

Because they don't have to worry about a LOCA going on concurrent with the loss of offsite power station blackout event. And that's probably the most important element.

And with that said, we also want to make sure we take credit for the PRA models that were modeled in the plant. As closely as built as operated as possible.

Plant's once they install the shutdown

seal will change their PRA models to credit the seal. It's a fairly simple change to the model.

Once we agree on the failure probabilities, it's an add onto the WOG2000 model, which the staff described. Or any other model that we may have.

We are actually looking at a possible change to the WOG2000 model that will address some things that aren't covered in that original model. But that's not part of this discussion.

But, we wanted to make sure the SE was written so we could -- that the report was written so we can take advantage of those future models as well.

And once again, once you integrate it into the PRA model, you do get a significant decrease in core damage frequency. And if we go the next. I'm actually just going to skip the next slide.

So, We're actually seeing over 50 percent reduction of core damage frequency. Typically on the order of 60 to 70 percent reduction.

Seal LOCAs have been probably the biggest thorn in our side as far as the PRA modeling, as far as our sequences that are concerned. So this really does provide us a tremendous benefit.

And with that, I'll open up for questions on the open session.

MEMBER SKILLMAN: Roy, why do you show PWRA and PWRB?

MR. LINTHICUM: Those are just -- those are two different plants.

MEMBER SKILLMAN: It just shows that for two different plants that they're P's -- there's a huge reduction.

MR. LINTHICUM: There's a huge reduction. Right. They're slightly different designs. But they're still -- there's still a huge reduction regardless of what type of plant.

If you have a Westinghouse RCP pump, regardless of whether or not you're a Westinghouse plant, there's actually some BNW plants with Westinghouse RCPs involved.

MEMBER SKILLMAN: Yes.

MR. LINTHICUM: Regardless of the plant design, you get a very large reduction.

MEMBER SKILLMAN: Okay. You're saying if it's a P and has a seal, that's what happens.

MR. LINTHICUM: That's what happens.

MEMBER SKILLMAN: Makes no difference whether it's a --

MR. LINTHICUM: Correct.

MEMBER SKILLMAN: BNW or a --

MR. LINTHICUM: Correct.

MEMBER SKILLMAN: Westinghouse origin or what's in origin.

MR. LINTHICUM: Yes. Well, I don't know of any combustion plants that have Westinghouse pumps that would have the seal design that are in the WOG2000.

But there are some that may have.

MEMBER SKILLMAN: Is it strongly influenced by whether it's a Byron Jackson pump or one of the other --

MR. LINTHICUM: This seal only applies to Westinghouse pumps.

MEMBER SKILLMAN: Only to Westinghouse pumps.

MR. LINTHICUM: Only Westinghouse pumps, right.

MEMBER SKILLMAN: Got it.

MR. LINTHICUM: The Byron Jackson pumps that are in like the CE plants, have a different seal design. And there's actually an approved topical on the PRA model for those seals as well.

MEMBER SKILLMAN: And the Bingham pumps?

MR. LINTHICUM: That's all -- all the CE

pumps are covered under that topical report on the CE plants.

MEMBER SKILLMAN: Thank you Roy.

MR. LINTHICUM: Okay.

MEMBER STETKAR: Anything else for the Owners Group in open session?

(No response)

MEMBER STETKAR: If not, are there any people in the room who would like to make a comment during the open session? If so, come up to the mic. And I'll ask if there are any members of the public on the bridge line who would like to make a comment. Please identify yourself and make a comment, please.

MR. BROWN: Bridge open.

MEMBER STETKAR: Thank you. Hearing none, We will now close the session and close the transcript. If you could bang the gavel.

(Whereupon, the above-entitled matter went off the record at 3:37 p.m.)

# ACRS Full Committee Meeting

# NRC Staff Review of Extended Power Uprate for Browns Ferry Nuclear Plant Units 1, 2, and 3

# Opening Remarks

## Kathryn Brock

**Deputy Director**

**Division of Operating Reactor Licensing**

**Office of Nuclear Reactor Regulation**

# Background

- **Browns Ferry Proposed EPU:**
  - ➢ **3458 to 3952 Megawatts Thermal**
  - ➢ **14.3% increase from Current Licensed Thermal Power**

- **EPU Review done using Extended Power Uprate Review Standard, RS-001**

- **No open items in draft safety evaluation**

# Introduction

## Farideh Saba

**Senior Project Manager**

**Division of Operating Reactor Licensing**

**Office of Nuclear Reactor Regulation**

# Review Schedule

- **September 21, 2015 – Application submitted to NRC.**
- **January 11, 2016 – Application accepted by NRC for review.**
- **May 3, 2017 – ACRS Subcommittee meeting**
- **NRC forecast for review completion July 2017**
- **TVA planned implementation dates:**
  **Unit 3 in Spring 2018**
  **Unit 1 in Fall 2018**
  **Unit 2 in Spring 2019**

# Agenda

## OPEN SESSIONS

- **TVA Presentations:**
  - **Background**
  - **Summary of Plant Modifications**
  - **Overview of Elimination of Credit for Containment Accident Pressure (CAP)**
  - **Overview of Replacement Steam Dryer**

## CLOSED SESSIONS

- **NRC staff Presentations:**
  - **Review of Fuel and Reactor System**
  - **Review of CAP Credit Elimination**
  - **Review of Replacement Steam Dryers**

# Browns Ferry Nuclear Plant
# Extended Power Uprate License Amendment Request
# Advisory Committee on Reactor Safeguards

June 7, 2017

# Introductions

## Browns Ferry Nuclear Plant
## Extended Power Uprate

Gerry Doyle
Director, Extended Power Uprate
and
Ed Schrull
TVA Corporate Fleet Licensing Manager

# BFN EPU ACRS – Introductions

## Agenda

| | |
|---|---|
| • Introductions | E. Schrull/G. Doyle |
| • Overview | |
| ➢ Background | S. Bono |
| ➢ Modifications Summary | L. Hughes |
| • Elimination of credit for Containment Accident Pressure (CAP) in Net Positive Suction Head (NPSH) evaluations for Emergency Core Cooling System (ECCS) pumps | P. Donahue |
| • Replacement Steam Dryer (RSD) Overview | D. Pappone |

# BFN EPU ACRS – Introductions

## Key Team Members Present

- Steve Bono – Browns Ferry Nuclear Plant (BFN) Site Vice President
- Lang Hughes – BFN General Manager, Site Operations
- Gerry Doyle – Director, Extended Power Uprate (EPU)
- Scott Hunnewell – BFN Director, Site Engineering
- Pete Donahue – EPU Senior Engineering Manager
- Denny Campbell – BFN Nuclear Plant Shift Operations Manager
- Bill Baker – EPU Senior Operations Manager
- Ed Schrull – Tennessee Valley Authority Corporate Fleet Licensing Manager
- Dan Green – EPU Senior Licensing Manager
- Michael Dick – EPU Project Engineer
- Greg Storey – EPU Project Engineer
- Jeff Lewis – EPU Project
- Bill Williamson – BFN Reactor Engineer

# BFN EPU ACRS – Introductions

Key Team Members Present (continued)

- Dan Pappone – General Electric – Hitachi (GEH)
- Don Sampson– GEH
- David McBurney – AREVA
- Alan Meginnis – AREVA

# BFN EPU ACRS Subcommittee – Introductions

## BFN EPU Team

- The EPU Project Team is staffed with personnel having extensive Boiling Water Reactor (BWR) plant and EPU experience
  - TVA
    - Combination of dedicated project and plant resources
  - GEH (Nuclear Steam Supply System)
  - AREVA (Fuels)
  - Industry EPU experienced specialty contractors

# BFN EPU ACRS Subcommittee – Introductions

BFN EPU Project Team Goals

- Present Nuclear Regulatory Commission (NRC) with a high quality License Amendment Request (LAR) consistent with RS-001 Standards
- Resolve Containment Accident Pressurization (CAP) credit issues
- Install Replacement Steam Dryers (RSD) for all three units
- Provide smooth transition to EPU for Plant Operations

# Overview

# Browns Ferry Nuclear Plant
# Extended Power Uprate

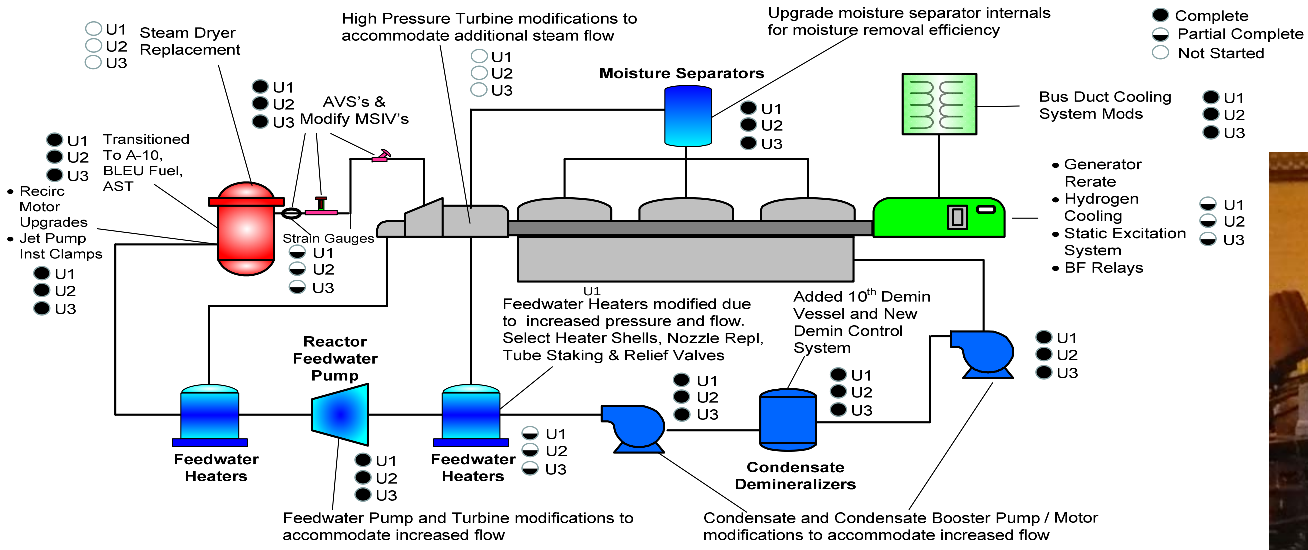
Steve Bono
Browns Ferry Nuclear Plant Site Vice President
and
Lang Hughes
Browns Ferry Nuclear Plant General Manager, Site Operations

# BFN EPU ACRS – Overview

## Background

- BFN
  - General Electric Boiling Water Reactor (BWR)/4 Mark I Containment
- Operating Licenses issued
  - Unit 1 - 12/20/1973, Unit 2 - 8/20/1974, Unit 3 - 8/18/1976
- Commercial Operation commenced
  - Unit 1 - 8/1/1974, Unit 2 - 3/1/1975, Unit 3 - 3/1/1977
- Renewed Licenses issued 5/4/2006 (Units 1, 2, and 3)
- Licensed Thermal Power History
  - Original Licensed Thermal Power (OLTP) - 3293 MWt
  - Stretch Uprate (105% OLTP) - 3458 MWt
    - Units 2 and 3 - 9/8/1998, Unit 1 - 3/6/2007
  - Proposed EPU (120% OLTP, 14.3% CLTP) - 3952 MWt

# BFN EPU ACRS – Overview Modifications Summary





Feed Water Pump


High Pressure Turbine


Condensate Pumps


Condensate Booster Pumps

# BFN EPU ACRS – Overview

## Modifications Summary

- Remaining Modifications
  - <u>Replacement Steam Dryer</u>
    - New steam dryers to be installed to increase structural design margin to accommodate EPU operation
  - <u>High Pressure Turbine Replacement</u>
    - High pressure rotors to be replaced with rotors designed for increase flow associated with EPU
  - <u>Feedwater Heaters</u>
    - Tube bundle and channel head to be replaced in the number 4 Feedwater Heaters with design less susceptible to flow-induced vibration
  - <u>Main Generator Hydrogen Pressure</u>
    - Main generator hydrogen pressure to be increased to support EPU operation

# BFN EPU ACRS – Overview

## Modifications Summary

- Remaining Modifications (continued)
  - Standby Liquid Control (SLC) System
    - Increase Boron-10 enrichment from 63 to 94 atom percent in SLC Storage Tank solution to lower power faster during Anticipated Transient Without Scram (ATWS)
  - Hardened Wetwell Vent (HWWV)
    - Modify HWWV, as part of modifications for compliance with the Order on Reliable Hardened Containment Vents, to provide capacity of HWWV that is one percent of EPU thermal power
  - Self-Excited Generator Excitation System
    - Modify excitation system to a self-excited shaft driven alternator and modify the Automatic Voltage Regulator to address a transient stability issue resulting from increased electrical generation at EPU conditions (North American Electric Reliability Corporation issue)
  - Alternate Leakage Treatment Pathway
    - Modify the Alternate Leakage Treatment Pathway design to be consistent with the existing licensing basis dose calculations performed at EPU conditions

# BFN EPU ACRS – Overview

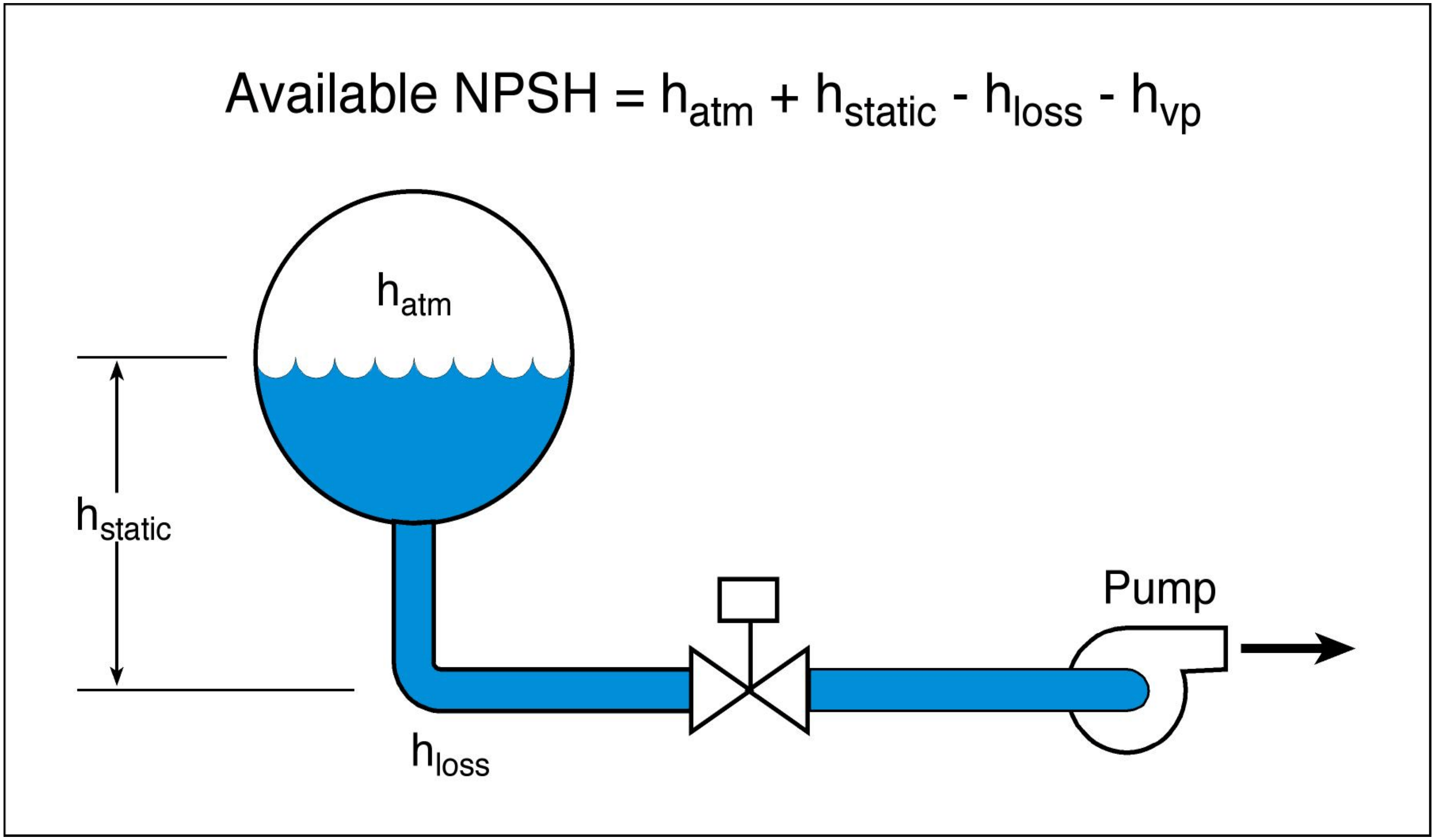
## Modifications Summary

- Remaining Instrumentation Changes to support EPU
  - Technical Specification Instrumentation Respan/Recalibration
    - Turbine first stage pressure scram bypass permissive setpoint to be recalibrated
    - Main steam line high flow isolation channel to be respanned
    - Average Power Range Monitor flow biased and setdown instrumentation to be respanned and setpoints recalibrated
  - Condenser Instrumentation Upgrade
    - Install nine new condenser vacuum pressure transmitters per unit (three on each condenser) and provide signals to electro-hydraulic control (EHC) system
    - Move condenser A/B/C low vacuum alarm, low vacuum turbine trip and low vacuum bypass trip functions to EHC logic
    - Perform hardware and software changes to EHC system to support alarm and trip functions

# Elimination of Credit for Containment Accident Pressure in Net Positive Suction Head Evaluations for Emergency Core Cooling System Pumps

Pete Donahue
Extended Power Uprate Senior Engineering Manager

# BFN EPU ACRS – Elimination of credit for CAP in NPSH Evaluations for ECCS pumps



# BFN EPU ACRS – Elimination of credit for CAP in NPSH Evaluations for ECCS pumps

## Background

- Opportunity to improve margins and address industry concerns associated with CAP credit
  - One of the EPU Project Team Goals
- Used guidance provided in SECY-11-0014, Use of Containment Accident Pressure in Analyzing Emergency Core Cooling System and Containment Heat Removal System Pump Performance in Postulated Accidents
- EPU NPSH evaluations for ECCS pumps demonstrate **credit for CAP is NOT required**

# BFN EPU ACRS – Elimination of credit for CAP in NPSH Evaluations for ECCS pumps

## Actions to Eliminate CAP Credit

- Performed comprehensive analyses of accidents and special events
- Increased Residual Heat Removal (RHR) Heat Exchanger factor (K-value) for all events
- Validated acceptability of increased RHR Heat Exchanger K-value by performing RHR Heat Exchanger Performance Tests on all 12 RHR Heat Exchangers at BFN (4 per unit)
  - To ensure that this level of RHR Heat Exchanger performance is maintained, a Technical Specification for the RHR Heat Exchanger Performance Monitoring Program will be added
- Increased SLC System Boron-10 enrichment for ATWS event
- Used Vendor supplied NPSH curves (NPSHr 3% curves)
- Used Technical Specification values for initial conditions in Design Basis Accident (DBA) Loss of Coolant Accident (LOCA) and Special Events analyses
  - Except for Fire Event
    - SECY-11-0014 allows use of nominal values for Special Events

# BFN EPU ACRS – Elimination of credit for CAP in NPSH Evaluations for ECCS pumps

Results

- For all events
  - Net Positive Suction Head Available (NPSHa) is greater than Effective Net Positive Suction Head Required ($NPSHr_{eff}$)

- **No CAP credit required**
  - No additional operator actions required

# Replacement Steam Dryer (RSD) Overview

Dan Pappone
GE – Hitachi Nuclear Energy

# BFN EPU ACRS – RSD Overview

Summary

- Based on the curved hood six bank prototype replacement dryer first used in a BWR/4 reactor
- Design is significantly more robust than the original steam dryer it replaces
- BFN acoustic load definition developed using Main Steam Line (MSL) acoustic pressure measurements taken at the three BFN units
- The Flow Induced Vibration (FIV) fatigue evaluation and primary stress methodologies used for BFN RSD analysis has been reviewed in detail on BWR/6 RSD and Economic Simplified Boiling Water Reactor projects
- Analyzed for the applicable primary structural loads for normal operation and for transient and accident conditions

# BFN EPU ACRS – RSD Overview

## EPU RSD Experience

- Successful operating history for RSDs at EPU conditions
  - Quad Cities Units 1 and 2 RSDs
  - Dresden Units 2 and 3 RSDs
  - Vermont Yankee original dryer with modifications
  - Susquehanna Units 1 and 2 RSDs
  - Grand Gulf RSD

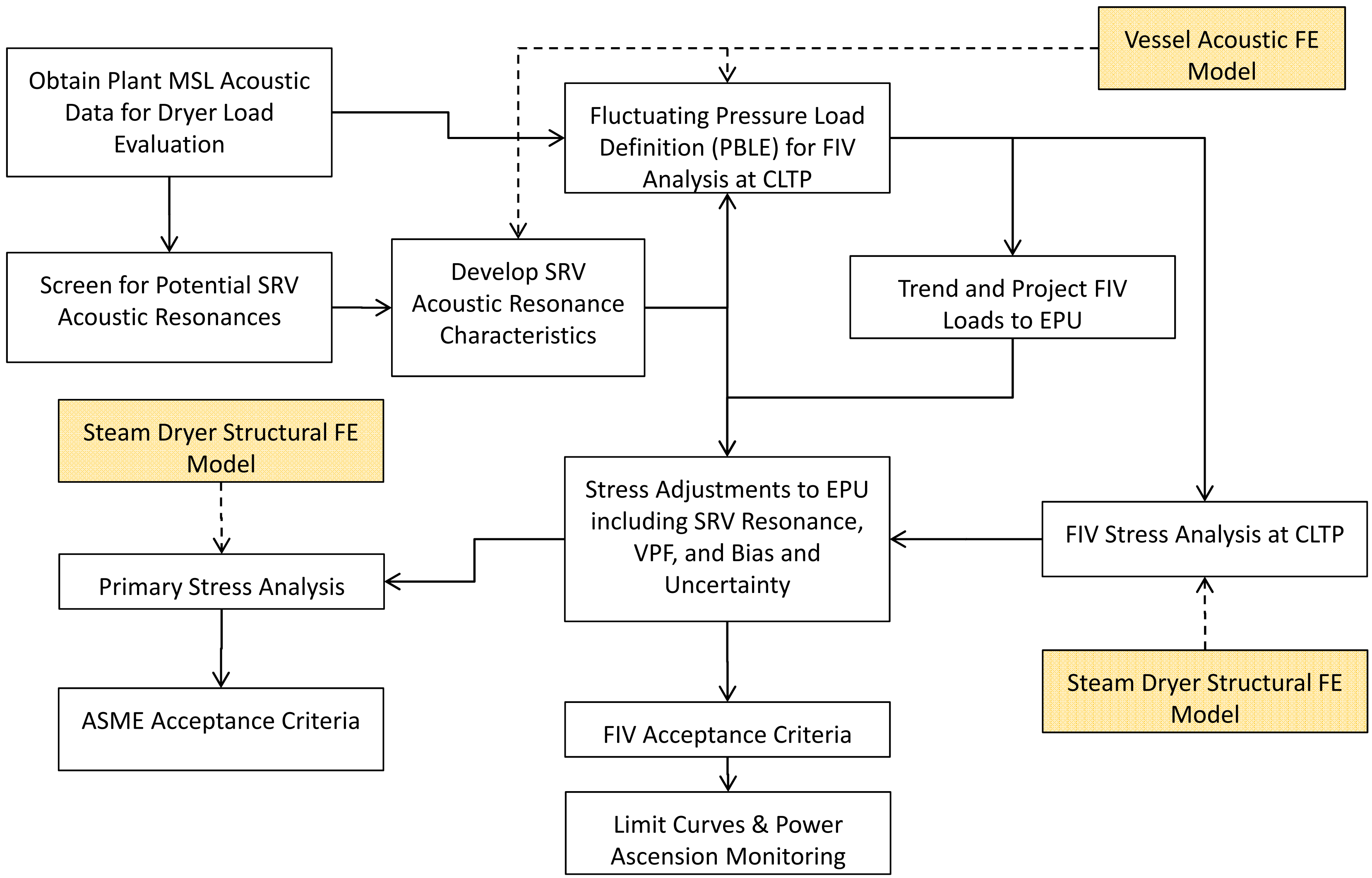
# BFN EPU ACRS – RSD Overview

## BFN RSD Design

- Changes to prototype dryer design
  - Dryer/vessel interface changes
  - Address Operating Experience lessons learned
  - Refine design details to reduce stresses

# BFN EPU ACRS – RSD Overview

## BFN RSD Analysis



# BFN EPU ACRS – RSD Overview

## BFN RSD FIV Analysis

- BFN RSD Finite Element Model (FEM)
  - Used to develop BFN-specific design improvements
  - FIV stress adjustments applied to FEM results

- BFN RSD Fatigue Analysis Results
  - BFN RSDs satisfy the NRC recommended minimum alternating stress ratio of 2.0 for all practical purposes

# BFN EPU ACRS – RSD Overview

## BFN RSD Primary Stress Evaluation

- RSD is a non-safety related item
  - Classified as an Internal Structure as defined in American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Subsection NG, Paragraph NG-1122
- RSD is not an ASME Code component
- BFN RSD Primary Stress Results
  - RSD was evaluated for Normal, Upset, Emergency and Faulted conditions
  - Stresses for each load combination were compared to ASME Code, Section III, NG stress limits for all RSD components
  - The stresses for all structural components are below the ASME Code allowable limits at EPU operating conditions

# BFN EPU ACRS – RSD Overview

## BFN RSD Power Ascension Monitoring

- Power ascension monitoring process similar to previous RSD projects
  - Addressed in BFN License Conditions
- BFN Lead Unit
  - Both RSD and MSLs instrumented
  - Monitor using strain gauges, on-dryer pressure transducers, and accelerometer
  - Confirmatory structural analysis for lead unit at EPU based on on-dryer measurements
- Follow-on Units
  - MSLs instrumented
  - Confirmatory structural evaluation based on MSL measurements

# BFN EPU ACRS – RSD Overview

## BFN RSD Power Ascension Monitoring (continued)

- BFN License Conditions address
  - BFN Lead Unit Test Plan, including hold points for NRC review of data and evaluations
  - BFN Follow-on Units Test Plan, including hold points for NRC review of data and evaluations
  - BFN RSD Inspection Plan consistent with industry guidance

# BFN EPU ACRS – Acronym List

- ASME – American Society of Mechanical Engineers
- AST – Alternative Source Term
- ATWS – Anticipated Transient Without Scram
- AVS – Acoustic Vibration Suppressors
- BFN – Browns Ferry Nuclear Plant
- BLEU – Blended Low Enriched Uranium
- BWR – Boiling Water Reactor
- CAP – Containment Accident Pressure
- CCW – Condenser Circulating Water
- CLTP – Current Licensed Thermal Power
- DBA – Design Basis Accident
- ECCS – Emergency Core Cooling System
- EHC – Electro-hydraulic Control
- EPU – Extended Power Uprate
- F – Fahrenheit
- FE – Finite Element
- FEM – Finite Element Model
- FIV – Flow Induced Vibration
- GEH – General Electric – Hitachi
- HWWV – Hardened Wetwell Vent
- LAR – License Amendment Request
- LOCA – Loss of Coolant Accident
- MSIVs – Main Steam Isolation Valves
- MSL – Main Steam Line
- MWt – Megawatts Thermal
- NPSH – Net Positive Suction Head
- NPSHa – Net Positive Suction Head Available
- $NPSHr_{eff}$ – Effective Net Positive Suction Head Required
- NRC – Nuclear Regulatory Commission
- OLTP – Original Licensed Thermal Power
- PBLE – Plant Based Load Evaluation
- RHR – Residual Heat Removal
- RSD – Replacement Steam Dryer
- SLC – Standby Liquid Control
- SRV – Safety Relief Valve
- VPF – Vane Passing Frequency

# BFN EPU ACRS – Questions/Comments