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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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640TH MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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OPEN SESSION

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THURSDAY

FEBRUARY 9, 2017

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ROCKVILLE, MARYLAND

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

COMMITTEE MEMBERS:

DENNIS C. BLEY, Chairman

MICHAEL L. CORRADINI, Vice Chairman

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RONALD G. BALLINGER, Member

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JOY REMPE, Member
GORDON R. SKILLMAN, Member
JOHN W. STETKAR, Member
MATTHEW W. SUNSERI, Member

DESIGNATED FEDERAL OFFICIALS:
CHRISTOPHER BROWN
DEREK WIDMAYER

ALSO PRESENT:
TONY AHN, KHNP
DENNIS ANDRUKAT, NRO
JOHN BUDZYNSKI, NRO
ALEXANDRA BURJA, NRO
CHANG SOK CHO, KEPCO NF
WOOCHONG CHON, KEPCO NF
IN-CHEOL CHU, KAERI
JEFF CIOCCO, NRO
ANTONIO DIAS, NRO
ADAKOU FOLI, NRR

JAMES GILMER, NRO
ZACHARY GRAN, NRO
SYED HAIDER, NRO
BRAD HARVEY, NRO
ALFRED HATHAWAY, RES
JERMAINE HEATH, NRO
DAVID HEETZEL, NRO
KAIHWA HSU, NRR
AMY HULL, RES
SEOK HWAN HUR, KEPCO E&C
HO CHEOL JANG, KEPCO E&C
HYGOK JEONG, KEPCO E&C
SUNG HWAN JUN, KEPCO E&C
JOSHUA KAIZER, NRR
KYUNG WANG KANG, KEPCO E&C
REBECCA KARAS, NRO
KERRI KAVANAGH, NRO
JONG SOO KIM, KEPCO E&C
JUNGHO KIM, KHNP
KANGHOON KIM, KEPCO NF
TAE HAN KIM, KEPCO
YUN HO KIM, KHNP
PAUL KROHN, NRO
YONGSUN LEE, KEPCO E&C
CHANG-YANG LI, NRO

DAE HEON LIM, KEPCO E&C

KWANGIL LIM, KEPCO E&C

SHANLAI LU, NRO

GREG MAKAR, NRO

STEVE MANNON, AECOM

MICHAEL D. MAZAIKA, NRO

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RICHARD MCINTYRE, NRO

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JOHN NAKOSKI, RES

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JIYONG OH, KHNP

NGOLA OTTO, NRR

SUNG KEW PARK, KEPCO NF

KEVIN QUINLAN, NRO

SHEILA RAY, NRR

ERIC REICHELT, NRO

TAE SUN RO, KEPCO E&C

JAMES ROSS, AECOM

TARUN ROY, NRO

DAVE RUDLAND, NRR

SUNG-JE SEO, KEPCO E&C

ROBERT SISK, Westinghouse

SWAGATA SOM, NRR

DUK-BIN SONG, KEPCO E&C

GARY STEVENS, Formerly NRC*
ANGELO STUBBS, NRO
JEONGKWAN SUH, KHNP
YIXING SUNG, Westinghouse
MATT THOMAS, NRO
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JOE WILLIAMS, NRO
STEVE WILLIAMS, NRO
GEORGE WUNDER, NRO
TAE YOUNG YOON, KEPCO NF

*Present via telephone

C-O-N-T-E-N-T-S

P R O C E E D I N G S

8:31 a.m.

CHAIRMAN BLEY: The meeting will now come to order. This is the first day of the 640th meeting of the Advisory Committee on Reactor Safeguards.

During today's meeting, the committee will consider the following: first, selected chapters and safety evaluation reports with open items associated with the Advanced Power Reactor 1400 design certification and selected topical reports. Second, Draft Final Reg Guide 1.207, Guidelines for Evaluating the Effects of Light-Water Reactor Coolant Environments and Fatigue Analyses of Metal Components. Third, Generic Quality Assurance Lessons Learned for New Reactors. And fourth, we will be working on ACRS reports.

The ACRS was established by statute and is governed by the Federal Advisory Committee Act, FACA. As such, this meeting is being conducted in accordance with the provisions of FACA. That means the committee can only speak through its published 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 also set aside 10 minutes for spur-of-the-moment comments from members of the public attending or listening to our meetings. Written comments are also welcome. Mr. Christopher Brown is the Designated Federal Official for the initial portion of this meeting.

A portion of the session on selected chapters of the SER with open items associated with the APR1400 design certification may be closed to protect proprietary information applicable to this matter. The ACRS section of the U.S. NRC public website provides our charter bylaws, letter reports, and full transcripts of all Full and subcommittee meetings, including all slides presented at the meetings. We have no written comments or requests to make oral statements from members of the public regarding today's sessions.

There will be a telephone bridge line. To preclude interruption of the meeting, the phone is 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 the speakers use one of the microphones, identify themselves, and speak with sufficient clarity and volume so that they can be readily heard.

I also want to make you aware that this meeting is being webcast, with the ability to view our presentation slides on the web. Anyone out there on the bridge line who may want to do that can dial into the bridge line or connect through the NRC's public meeting website and click on the link there. It does work. If it does not, please call our office. And also, the -- the sound quality on that website is usually better than on the phone line.

At this time, I am going to turn the meeting over to Member Ron Ballinger. Professor Ballinger?

MEMBER BALLINGER: Thank you, Chairman.

(Pause.)

MEMBER BALLINGER: Thank you, Mr. Chairman. The APR1400 Subcommittee has been reviewing the DCD and SERs with open items for the past several months. We have had a -- I think about six meetings related to Chapters 2, 5, 8, 10, and 11, and this meeting is going to be -- KHNP and the staff will be reporting on the results of those meetings.

And also, we have also reviewed two topical reports, the CHF Correlation and the Fluidic Device. I would call people's attention to item number six on the agenda, which will be a closed session, and -- and item number seven, the -- KHNP presentation on the CHF Correlation will be closed, and the Fluidic Device will be open. So when we get to that point, I think it is after a break, we will have to make some adjustments to the -- to the system.

But it has been kind of a long -- longer, intense review period, and the subcommittee appreciates the effort that -- that both the staff and KHNP has put into this effort. And now I would like to turn it over to Jeff Ciocco --

MR. CIOCCO: Thank --

MEMBER BALLINGER: -- for --

MR. CIOCCO: -- you --

MEMBER BALLINGER: Yes.

MR. CIOCCO: -- Dr. Ballinger. My name is Jeff Ciocco. I am the Lead Project Manager for the APR1400 Standard Design Certification Project. Along with many of our technical staff in the audience, we thank you for -- for having us down here to the -- for the Full Committee. This is a significant milestone in our -- in our project after presenting to the subcommittees Chapters 2, 5, 8, 10, and 11, as well as the two topical reports, so thank you for having us.

MEMBER BALLINGER: And I guess now we just

-- the floor is yours.

MR. Y. KIM: Yes, good morning. I am Yun Ho Kim.

Good morning. I am Yun Ho Kim. I am Deputy Project Manager from KHNP. I appreciate for your sharing time with APR1400, this application review, so I think that I am ready to present for the Chapter 2, 5, 8, 10, 11. So I will start?

MEMBER BALLINGER: Yes.

MR. Y. KIM: Yes. First, I want to say APR1400 is essentially complete design. The APR1400 reference plan, singly Unit 3 went into commercial appraisal at December last year, and the -- and that the construction of Barakah plant is going on on schedule, so we think APR1400 is essentially complete design.

So first, let me brief the -- the distinguishing design features for APR1400. First, we used the fluidic device in our safety injection system, and the over-pressure protection. We used POSRV, and also we -- we used the reflective metal insulator for addressing the GSI-191, and we also used seismic design with finite element model method, and for the enhanced SBO coping capability, we used the gas turbine for alternate AC source, and for the battery capacity, we -- we our capacity is for 16-hour capacity.

Also, we have improved design -- design for tolerance to beyond design basis, like such as aircraft impact analysis by 10 CFR Part 50.150. Also, we adopted the loss of large area design and the physical security design. Next.

For Chapter 2, Chapter 2 describes the site interface requirements for APR1400 design, including geological, seismological, hydrological, and meteorological characteristics. In our DCD table, Table 2.0-1, presents the site-related parameters, such as maximum elevation of groundwater and design temperature and the seismic information for the APR1400. So the combined license applicant is to confirm the site characteristics are bounded by the parameters in our Table 2.0-1. Next.

In Chapter 2, we also describe atmospheric dispersion factors for dose evaluations during normal and extraordinary condition, for long-term X/Q, D/Q, to calculate the normal offsite dose due to the gaseous release, and short-term X/Q to evaluate the radiological consequence analysis for design basis accident as specified.

The 95th percentile onsite atmospheric

dispersion factor was used for our MCR and TSC habitability analysis, and the data is presented in our DCD Table 2.3-2 through 12. Next.

The horizontal and the vertical safe shutdown earthquake are developed from the NRC Reg Guide 1.67 response spectra. That is anchored to peak ground acceleration value 0.3g. The safe shutdown earthquake in APR1400 design used the certified seismic design response spectra that is drawn in this figure. So overall, I think that in Chapter 2, we have no special technical issue in here. This is end of my Chapter 2. Can I now move onto next chapter?

Yes, for Chapter 5, for the reactor coolant system and the connecting system, reactor coolant system and the connecting system, the main picture is shown in the slide. The design life is 60 years. Electrical power is 1400 MWe, and that is according to two steam generator, four pump, and the one pressurizer.

One of the major differences from this -- our design and from systematic processes is that we adopt the POSRV. That is used for our over-pressure protection. Let's move next, please.

For over-pressure protection, there are four sets of POSRV in the pressurizer, and there are two spring-loaded pilots for each POSRV. These pilot valve open the main valve, would be the high system pressure. Both motor-operated pilot valve are in series to manually open the main valve for rapid depressurizing of reactor coolant system. LTOP valves are also provided in the shutdown cooling system suction line for over-pressure protection when reactor coolant system is at low temperature. Also, we use main safety valve, also provides secondary site of over-pressure protection. Next.

For the materials side of reactor coolant system, APR1400 design used proven material having successful operating experience and that met code and licensing requirement. They are compatible to reactor coolant and resistant to various degradations such as corrosion, stress corrosion cracking, fatigue, and neutron radiation effects.

In this reactor coolant component, there are three major material used. That is low-alloy steel, Alloy 690, and Austenitic stainless steel. Low-allow steel is just mainly for the main component and the main nozzle. That is collected with Austenitic stainless steel or nickel-based alloy. And Alloy 690 is used for steam generator tube, CEDM, and ICI nozzle and small, less than one inch small nozzles. For Austenitic stainless steel, that is mainly used for our internal and core support structure.

So in overall, Chapter 5, also we think that there is no critical issue.

MEMBER STETKAR: I have a question on -- we had some discussion during the subcommittee meeting regarding the configuration of your containment spray and shutdown cooling systems. The design uses a containment spray pump and a shutdown cooling pump interchangeably. In other words, they are identical pumps. You can use either for each function.

And I have to apologize to everyone because I need to get into a little bit of piping details here. The intent is that for example if I remove a containment spray pump from service for maintenance that the shutdown cooling pump in that division will be aligned so that it can take suction from the in-containment refueling water storage tank, the IRWST, and have its discharge aligned to the containment spray header, and that in that alignment, the shutdown cooling pump will start automatically from a containment spray signal. Is that correct? I want to make sure that I have the design philosophy correct.

MR. Y. KIM: Right.

MEMBER STETKAR: Okay. When we looked at the piping diagrams in the design certification document, I didn't find a local suction isolation valve for the containment spray pumps, so that when I remove a containment spray pump from service, according to the information that we had, it was necessary to physically close the IRWST suction valve that would be used to supply the shutdown cooling pump.

With that valve closed, the shutdown cooling pump cannot take suction from the IRWST, and the position of that valve provides the automatic starting interlock for the shutdown cooling pump. Therefore, it is not clear to me how the shutdown cooling pump can be used as a replacement for the containment spray pump. So could you comment on that?

At the -- at the end of the subcommittee meeting, we left it that KHNP would get back to us with information about whether there is a local suction isolation valve for the containment spray pump that could be closed when that pump is removed from service, and we have not received any feedback regarding that. So is there?

MR. T. KIM: This is Tae Han Kim from

KEPCO E&C. This item we provide at the follow-up presentation for Chapter 5, question 11. The CS pump suction from the IRWST should be closed to perform the shutdown cooling function using -- CS Pump 347 should be closed. This one we provide at the follow-up presentation.

MEMBER STETKAR: The -- the valve that I am concerned about is -- I know. I've got that follow-up presentation, and the valve that I am concerned about in particular on Division 1 would be Valve 340. 347 also needs to be closed, and that is an additional valve, so what I am hearing -- and I want to make sure that I -- that I understand this -- is that there is no manual suction valve for the containment spray pump, and I think that was confirmed during that follow-up discussion.

MR. T. KIM: Yes.

MEMBER STETKAR: Okay. I just wanted to make sure we had that on the record because I didn't know whether there was additional information that wanted to be presented today. Thank you.

MR. Y. KIM: Yes. Then -- then let me move on to Chapter 8, Electric Power System.

One thing on the picture of APR1400 electric power system is we adopt the four train design. The Class 1E onsite AC&DC power system consists of four independent trains. Each redundant load group consists of divisional pair of trains: Train A plus C is Division 1; Train B plus Train D is Division 2.

Each train has the designated EDG it is assigned, and each train is physically and electrically independent of non-Class 1E power system and other Class 1E trains. There is no interconnection and load share between trains. Class 1E equipment of each train is located in the dedicated locations, so we arranged in quadrant division. Next.

For the SBO coping capability, we -- we use 16-hour SBO coping duration that is considered in our design. The type of AAC source is diversified from emergency AAC source. The AAC GTG gas turbine generator has sufficient capability and capacity and reliability to bring the plant into safe shutdown condition and will be aligned to a shutdown bus without ten minutes at onset of SBO. 16-hour duty cycle is considered for DC trains, trains' C and D batteries to support C and D plant equipment, such as turbine-driven aux feedwater pump, during SBO.

In Chapter 8, we -- we think that there is two issues. We -- actually, we have five open items from this, but in terms of technical view, we think two technical issues in here. One is open phase conditions. The other is compliance with SECY-91-078.

Concerning the open phase issue, KHNP provided the response to RAI 85 -- 8521, including the design vulnerability study and a set of DCD markups that incorporated the design features of open phase detection system and the necessary COL items and ITAAC.

For SECY-91-078 issue, KHNP provided response to RAI 8426, including appropriate explanation and the justification as to how the APR1400 offsite power system design satisfies the GDC 17 and SECY-91-078 requirement in detailed manner.

For both issues, KHNP is waiting for staff's feedback. That is the current situation for Chapter 8.

Let's move on, Chapter 10. Chapter 10 is the -- the Steam and Power Conversion System. It converts heat generated by the reactor to -- to the electrical energy by using condensing cycle. Turbine generator system consists of 1800 rpm, and one is -- one HP, high-pressure, turbine, and the three low-pressure turbines, and the moisture-separating reheater, and the exciter, controls, et cetera. Next page.

For main steam system that has -- for main steam line with two steam generator and one dump valve -- steam dump valve, five safety valves, one isolation valve, and one isolation bypass valve. The -- the acronym for it is in the last page, so -- .

Also, one main steam line of each steam generator supplies steam to the associated turbine-driven aux feedwater pump. Turbine bypass system has the capacity to bypass 55 percent of steam to condenser through the eight turbine bypass valves.

In the aux feedwater system, each division has 100 percent motor-driven pump and 100 percent turbine-driven pump, and one storage tank.

For the materials side, steam and the feedwater system use CrMo alloy steel to prevent flow-accelerated corrosion, and we also use carbon steel with some additional thickness adoption.

So there is -- we think that there is no special issue for Chapter --

MEMBER BALLINGER: So --

MR. Y. KIM: -- to.

MEMBER BALLINGER: -- to clarify the carbon steel thickness, what you're saying is that you have -- you assume that a FAC could occur, or corrosion, and then what you have done is to just allow for an additional corrosion allowance --

MR. Y. KIM: Yes.

MEMBER BALLINGER: -- to make sure that --

MR. Y. KIM: Yes, just for all additional corrosion allowance --

MEMBER BALLINGER: Now --

MR. Y. KIM: -- so yes.

MEMBER BALLINGER: -- will that be verified with an inspection system going forward? In other words, to verify in fact that you have got the additional corrosion allowance that you need, since now you are assuming that you're going to get flow-assisted corrosion, so there will be a program established for inspection -- piping inspection?

MR. Y. KIM: Yes. Let me confirm that from our technical staff.

MR. SEO: This is Sung-Je Seo, KEPCO E&C, Mechanical Engineer. The corrosion allowance is based on the OPR1000 FAC, so there are two types of criteria. Additional thickness for -- for steam piping is 0.03, empty. For water piping, additional thickness is 0.06, and it is considered. Therefore, these criteria is also considered APR1400.

MEMBER BALLINGER: I guess the concern

that I had was that I will give you that there is not much difference between OPR1000 and APR1400, and there is plenty of data to support that, but the nature of FAC is such that it is not necessarily linear, and that just because it is good for OPR1000 does not mean that the piping configuration itself for the APR1400 might be slightly different, which would then result in a different kind of assessment for FAC, which would, you know, make the additional flow-assisted corrosion allowance insufficient.

MR. Y. KIM: I see, yes.

MEMBER STETKAR: Mr. Kim, I had a couple of questions on this. The APR1400 is designed to accept a full load rejection without a reactor trip, is that correct? So that you have a combination of steam relief capacity and the reactor power cutback system is designed to maintain the reactor operating after full load rejection, and that is one of the reasons why you have 55 percent turbine bypass capacity.

I didn't have it in my notes, and I can't find it: do the MSADVs, the atmospheric dump valves, receive automatic signals to close, or are they only operated manually? I am sorry, not to close, to open.

MR. SEO: Yes. This is also Sung-Je Seo.

MSADV is a manual operation.

MEMBER STETKAR: Only manual, okay.

MR. SEO: Yes.

MEMBER STETKAR: So -- so for the runback, the power cutback from a full load rejection, do you challenge the main steam safety valves to open?

MR. SEO: No.

MEMBER STETKAR: No, so the power cutback plus the turbine bypass valves should handle the load rejection without even challenging the main steam safety valves?

MR. SEO: That is right.

MEMBER STETKAR: Okay. And therefore you should have no challenges either to the pressurizer POSRVs, is that correct?

MR. SEO: Yes, right.

MEMBER STETKAR: So the -- okay. I just wanted to make sure I understood that, that you handle the full loadback with a reactor cutback and the turbine bypass valves.

MR. SEO: Yes.

MEMBER STETKAR: Thank you. That was information, I just wanted to make sure I had that clear in my own mind. Thank you.

MR. Y. KIM: Lastly, Chapter 11, for

radioactive waste management. Chapter 11, radioactive source terms, we have four categories: design basis source terms, expected source terms, and the secondary system activity, and the rad waste system source term that is described in our DCD.

These source terms are used for design of radioactive waste management system and for determining the equipment internal dose and to estimate the annual effluent release to the environment.

Rad waste management system consists of liquid, gaseous, and solid waste management systems. Liquid waste management systems use reverse osmosis technology pretreatment to remove organic matters and ion exchangers to remove specific items.

Gaseous waste management systems use the charcoal delay beds to delay xenon. The solid waste management system uses spent resin-drying system, long-term storage tank, and the solid waste compactor and filter-handling system. Next.

Process and effluent radiation monitoring and sampling system, PERMSS, measures and records the radioactivity level of liquid and gaseous process streams and effluents from liquid, gaseous, and other process systems during normal operation and AOO and the postulated accident. PERMSS for APR1400 design uses this list of monitors. I will skip to mention all of them here.

Offsite public maximum organ dose to -- due to normal operation are estimated for gas and liquid release. These -- these are set by the Part 50 Appendix I limit, and design basis effluent concentration exclusion area boundaries are within 18 percent of effluent concentrated limit of Part 20 Appendix B, 16.2 percent for gaseous effluents.

Liquid waste management -- waste system failure analysis shows that boric acid storage tank failure is the worst-case accident. Because of absence of site-specific information for groundwater, minimum required dilution factor was evaluated, and the result can meet the concentration limit of potable water in Part 20 Appendix B. There was estimated to be 9340.

For gaseous rad waste system failure analysis, to maximize the consequences of accident, inadvertent bypass of charcoal delay beds in gas waste management system is assumed. Estimated doses at EAB and LPZ meet the acceptance criteria of 10 millirem specified in SRP.

So there are currently three open items

associated with Chapter 11 that is under analysis evaluation now, so this is my end of presentation for Chapter 11. Thanks.

MEMBER BALLINGER: Questions? Questions? I have to say everything twice, it appears.

(No audible response.)

MEMBER BALLINGER: If there aren't any questions, then we can swap out with -- thank you very much.

MR. Y. KIM: Thank you.

MEMBER BALLINGER: And get the staff in.

MR. CIOCCO: Ready? Okay. Thank you, and good morning, everybody. I am Jeff Ciocco. I am the Lead Project Manager for the APR1400 Standard Design Certification project. I am just going to do the first two overview slides, and then turn it over to our -- to our Chapter Project Manager.

Slide 2, this is our 42-month review schedule parsed into a six-phase review, plus the rulemaking makes it about a 50-month project total. In Phase 1, we issued RAIs and the internal preliminary safety evaluation reports, and that has been completed. Phase 2, we issued a safety evaluation report with open items, and that is currently underway here.

Phase 3 is when the ACRS subcommittee and Full Committee reviews our safety evaluation report with open items, and that is currently underway as well. Phase 4, we worked a handful of issues remaining to closure and then issued the advanced safety evaluation report with open items -- or, I am sorry, with no open items, and that is currently underway as well for those chapters that have been completed in Phase 2.

In Phase 5, we will come back here and tell you how we resolved the final issues of the project. In Phase 6, we will issue the final safety evaluation report with no open items, and then we will complete the project with the Part 52 -- Part 52 rulemaking.

So Phases 2, 3, and 4 are currently underway in parallel, and what you hear and see now are the results of Phase 2, the safety evaluation report with open items. The safety evaluations are the result of huge staff efforts producing what will be about a 3500-page safety evaluation report where we combine all the total 23 chapters together, and I assure you we are working assiduously and enthusiastically to issue quality safety -- quality safety evaluation reports on time.

And I must say, maintaining the subcommittee's schedule is a priority for us. We try hard not to miss submission deadlines to the ACRS, but when we do, we certainly appreciate your efforts and Chris Brown's to -- to reschedule the subcommittee meetings to help us maintain the overall 42-month schedule for Phases 1-6. Next?

So my kind of summary here, the -- the staff has issued seven safety evaluation reports so far with open items for Chapters 2, 4, 5, 8, 10, 11, and recently, Chapter 12, which is not included in that -- in that bullet. Of the seven chapters issued, all but Chapters 4 and 12 have been -- as of today will have been presented to the -- to the Full -- to the Full Committee.

The staff has also issued safety evaluation reports with no issues on three of our five topical reports. The first is a quality assurance program description document, which is an improved document, and the other two, you are going to hear following us today -- or following the chapter presentation. One is on the critical heat flux, and the other is on the fluidic device.

So that is -- that is all I have for the overview, so I will now turn it over to our Chapter

Project Managers.

MR. ROY: Good morning. My name is Tarun Roy. I am the Project Manager for coordinating the staff review of APR1400 Chapter 2, Design Certification Application, and also Chapter 9 and 11.

Chapter 2 is the site characteristics. The technical topics of interest is, as the following information is site-specific, the combined operating license COL applicant is required to provide the site-specific information. Section 2.1 is the Geography and Demography. The COL applicant is to provide the site-specific information as part of COL information item 2.1(1) in the COL application.

Section 2.2, Nearby Industrial, Transportation, and Military Facilities, the COL applicant is to provide the site-specific information as a part of COL information item 2.2(1) and COL information item 2.2(2) in the COL application.

Section 2.3 is Meteorology. The SER for Section 2.3 addresses the regional climatology, local meteorology, onsite meteorological measurements program, short-term atmospheric dispersion estimates for accident releases, and long-term atmospheric dispersion estimates for routine releases. And staff reviewed the adequacy of the DCD site parameters related to regional climatology, short-term atmospheric dispersion estimates, and long-term atmospheric dispersion and deposition estimates.

The COL applicant is to perform the radiological consequences analyses to demonstrate that the related dose limits specified in 10 CFR 50.34 and GDC 19 are not exceeded if the site-specific X/Q values exceed the bounding values described in Table 2.3-1 and 2.3-12 of the FSAR.

All regulatory requirements for Section 2.3 have been satisfied.

2.4, Hydrologic Engineering: the SER for Section 2.4 addresses hydrological description, floods, probable maximum flood on streams and rivers, potential dam failures, probable maximum surge and seiche flooding, probable maximum tsunami flooding, ice effects, cooling water channels and reservoirs, channel diversion, flooding protection requirements, low considerations -- low water considerations, groundwater, accidental release of liquid effluents in ground and surface water, and technical specifications and emergency operations requirements.

All regulatory requirements for Section 2.4 have been satisfied.

Section 2.5, Geology, Seismology, and

Geotechnical Engineering: the SER for Section 2.5 addresses geologic, seismological, geotechnical site parameters used for APR1400 structural design and analysis. Applicant properly specified appropriate geologic, seismologic, and geotechnical site parameters.

All regulatory requirements for Section 2.5 have been satisfied. So there is no open item for Chapter 2 as such. Thanks.

MS. UMANA: Good morning. My name is Jessica Umana. I don't have a microphone. I do? Sorry.

My name is Jessica Umana. I am the Chapter 5 PM for APR1400. I will be presenting the staff's review, their findings, and the remaining issues for their reactor coolant system and related systems.

The staff's area of review for this chapter covered the reactor coolant system, including the reactor vessel, steam generators, reactor coolant pumps, pressurizer, and associated piping. Most of the regulatory requirements for Chapter 5 have been satisfied. We do have just a handful -- less than a handful of remaining issues, and as of yesterday, I was informed that the shutdown cooling issue has been resolved. The applicant submitted a response, and the staff found it acceptable.

For the reactor coolant pressure boundary, the issue there is a simple issue of removing language that was improperly included in the DCD. It's a little too restrictive, and the staff does not really --

MEMBER STETKAR: Jessica, could you --

MS. UMANA: Yes.

MEMBER STETKAR: -- explain to me how the shutdown cooling issue has been resolved? Because I don't -- I didn't know that, so how -- the -- the issue that I raised in the -- my question to the applicant is that I don't understand how a shutdown cooling pump can be used to replace a containment spray pump that is removed from service such that the shutdown cooling pump can take suction from the IRWST and receive an automatic containment spray actuation signal to start for containment spray, given what we know about the configuration of the flow paths in the suction -- in those suction lines.

And in particular, if you want to be very, very specific, it is my understanding from everything that I have heard and read that shutdown cooling pump number one receives an automatic signal to start from containment spray actuation if and only if suction valve 340 is open. That suction valve provides the flow path to that pump from the IRWST. That is my understanding of how the wiring is and the flow path.

It is also my understanding that if a containment -- if containment spray pump number one is removed from service with its suction lines isolated, valve 340 must be closed. If valve 340 is closed, shutdown cooling pump number one does not get an automatic signal to start, nor is its flow path aligned to the IRWST. That is my question. Now that's a -- that's a convoluted discussion, but I want to make sure it's on the record, and I believe I spoke correctly with all of the right valve numbers and signals, but I want to make sure it is clear on the record, and I don't know how that concern has been resolved, so I -- could you elaborate please?

MR. DIAS: Hi. Let me intervene here. This is Antonio Dias. I am the Branch Chief of the Plant Systems in NRO, and John Stetkar is correct. There was a conversation with him yesterday, and we agreed that it is really not completely closed, okay, as you heard him saying before. Unfortunately, I did not get the chance to modify the slides for this, so I take responsibility on this, and what John states is correct, okay?

MEMBER STETKAR: Thanks, Antonio. I just --

MR. DIAS: Thank you.

MEMBER STETKAR: -- wanted to make sure that there was --

MR. DIAS: No --

MEMBER STETKAR: -- I remember --

MR. DIAS: -- I --

MEMBER STETKAR: -- no, I remember the conversation. I just -- this has been a fluid kind of discussion since the subcommittee meeting, and --

MS. UMANA: I would say that my slides have been pretty fluid too.

MEMBER STETKAR: Yes, no, that is --

MS. UMANA: There are some changes that have happened since.

MEMBER STETKAR: And the other thing, the other reason why I wanted to get it on the record is that any conversations that we have as individual members are really irrelevant for the purposes of the committee, so I just wanted to make sure that at 9:15 in the morning in front of the Full Committee, things hadn't evolved since -- since yesterday, so thanks a lot, Antonio.

MR. CIOCCO: Thank you.

MEMBER BALLINGER: I appreciate that.

MR. CIOCCO: So it's still an open item, and we'll come back in Phase 5 and let you know how it gets resolved.

MS. UMANA: Well, the issue I was referencing was something not related to that, so --

MEMBER STETKAR: Okay.

MS. UMANA: -- it was --

MEMBER STETKAR: I am --

MS. UMANA: -- minor.

MEMBER STETKAR: -- I am sorry.

(Laughter.)

MEMBER STETKAR: I am really sensitive to shutdown, the word shutdown cooling.

MS. UMANA: No, it was something not related to that at all, so -- and like I said, the development of the slides, while I thought they were done, it has just kind of been fluid, so I am glad that that has been clarified.

Moving on --

MEMBER SKILLMAN: Before --

MS. UMANA: I am sorry.

MEMBER SKILLMAN: Before you go on to eight, which is your next slide, let me ask a question. In our discussions yesterday, we were talking about Chapter 4, and we got onto the topic of bypass flow, core bypass flow. And the core bypass flow is really established by the fit-up in Chapter 5 of the reactor vessel inside ledge on the hot leg to the mating surface on the core support barrel.

And what happens is the reactor coolant system heats. The core barrel grows into the t-hot and closes a very large opening on both the hot legs, thus precluding or minimizing core bypass flow. And my question is to what extent did you review that design feature in Chapter 5? Because Chapter 5 includes the reactor vessel and the internals. It bears directly on the assumed 3 percent bypass flow.

MR. CIOCCO: We'll have to see if we have the Chapter 5 reviewer here for that --

MEMBER SKILLMAN: Thank you.

MR. CIOCCO: -- area.

MEMBER SKILLMAN: Answer later is fine. I am raising the issue --

MR. CIOCCO: Yes.

MEMBER SKILLMAN: -- that is a critical dimension. It is final fit-up. It is major just to the fraction of a millimeter. If that dimension is off, then your bypass flow is great.

MR. CIOCCO: Yes, we can look into that. I --

MEMBER SKILLMAN: Thank you.

MR. CIOCCO: -- I don't see the reviewer right now. If we can move on to Chapter 8, and we'll see if the reviewer --

MEMBER SKILLMAN: Yes sir.

MR. CIOCCO: -- gets here. We can answer now or --

MEMBER SKILLMAN: Thank you, Jeff.

MR. CIOCCO: -- later. You're welcome.

MEMBER SKILLMAN: Okay.

MS. UMANA: Okay. Am I good to --

MEMBER STETKAR: Dick, does Chapter 5 actually cover the reactor vessel internals, or just the reactor coolant system and the vessel itself?

MEMBER SKILLMAN: We just went into Chapter 5. Critical parameters --

MEMBER STETKAR: Turn your --

MEMBER SKILLMAN: -- critical parameters for flow are identified in the internals as part --

MEMBER STETKAR: Okay.

MEMBER SKILLMAN: -- of the reactor vessel --

MEMBER STETKAR: Okay.

MEMBER SKILLMAN: -- are identified.

MEMBER STETKAR: Okay.

MEMBER SKILLMAN: Yes sir.

MEMBER STETKAR: Thanks.

MS. UMANA: Okay. That's it.

MR. CIOCCO: Hold on. Shanlai, do you want --

MR. LU: Shanlai Lu from staff. I think that it's -- in terms of the gap, whatever the bypass flow fraction, and we talked about it, discussed this issue with the subcommittee yesterday, and the -- it is being reviewed as part of the Chapter 4 for Section 4.4 as a part of the impact on the total core flow and the DNBR margin, so that is part of the -- what do we call it, the CPC set point methodology, that is considering that one. So staff actually, from our perspective, we consider it as part of the uncertainty already considered as part of the methodology.

MEMBER SKILLMAN: I understood that from yesterday.

MR. LU: Right.

MEMBER SKILLMAN: But that is a different issue than I am raising.

MR. LU: Okay.

MEMBER SKILLMAN: I accept the 2.4 percent plus 0.6 percent for a 3 percent --

MR. LU: Right.

MEMBER SKILLMAN: -- bypass flow. That is dandy, but that is not the issue.

MR. LU: Okay.

MEMBER SKILLMAN: The issue is there is a manufacturing tolerance that actually trumps that number, and if the manufacturing tolerance is tight enough, then I believe we can have high confidence that the 3 percent is maximum. But if that gap is not controlled with great precision, then I think the 3 percent number is in question, and that is why I am asking the question.

MR. LU: Okay. I --

MEMBER SKILLMAN: I --

MR. LU: -- I understand. I think that that is the part that is related to the reactor vessel internals, right? It is not --

MEMBER SKILLMAN: It --

(Simultaneous speaking.)

MEMBER SKILLMAN: -- it is related to --

MR. LU: Right.

MEMBER SKILLMAN: -- two major components. It is related to the final manufacturing machine fit-out on the reactor vessel ID --

MR. LU: Right.

MEMBER SKILLMAN: -- face and the OD on the core barrel.

MR. LU: Okay. So your issues really relate to the manufacturing tolerance?

MEMBER SKILLMAN: On Chapter 5 for the reactor --

MR. LU: All right.

MEMBER SKILLMAN: -- vessel --

MR. LU: We'll --

MEMBER SKILLMAN: -- and the core support barrel.

MR. LU: We'll try to find more --

MEMBER SKILLMAN: Okay.

MR. LU: -- people to address that issue.

MS. UMANA: Okay.

MEMBER SKILLMAN: Have I made clear the question that I am asking?

MS. UMANA: Yes.

MEMBER SKILLMAN: Thank you.

MS. UMANA: Okay. I do have the last bullet to cover on the reactor coolant pump flywheel integrity. The issues included the applicant use of ultimate strength in lieu of yield strength in the design analysis, lack of analysis for the flywheel hub, and inspection criteria for the flywheel hub. The applicant has provided a reasonable approach in resolving these issues, and the staff is now waiting for confirmation. They want to see that information included in the analysis report.

And that is all I have for Chapter 5. Moving on to Chapter 8 next.

MR. WUNDER: Thank you, Jessica. Good morning. I am George Wunder. I am the Project Manager for Chapters 8 and 10, and I will be presenting those to you today. We will start with Chapter 8, Electric Power System.

The staff safety evaluation for Chapter 8 addressed the offsite power system, the onsite AC system, onsite DC system, and station blackout. The staff determined with the exception of two unresolved issues all applicable regulatory requirements have been met.

The first issue relates to the staff's concern that the applicant did not conform to the guidance in SECY-91-078, and therefore did not meet the Commission's expectations for new reactors for meeting GDC 17. The second relates to the applicant's addressing the bullet in 2012-01 for open phase detection and alarm. Next slide, please.

The Commission approved SECY-91-078 to assist the staff in assessing the means by which new reactor applicants meet various regulatory criteria. There are two big pieces of guidance in -- in the SECY, and the first is that the Commission wanted to be -- wanted the design to be such that an ultimate source of AC power will be available for non-safety loads. The APR1400 design allows for -- for power to be transferred to an alternate power supply in the event of the unavailability of the preferred power supply, so that piece of guidance is satisfied.

The second is that the Commission wanted the design to be such that at least one offsite power source is available to each Class 1E bus, with no intervening non-safety bus, and it is in this area that we found something that needed a little bit of work. The staff noted that the APR1400 has a configuration in which both the standby auxiliary transformers and the unit auxiliary transformers have secondary windings that serve both safety and non-safety loads. In the staff's mind, this left the potential for failures on the non-safety side to impact safety side, since no analysis demonstrating that this could not happen had been provided. Next slide, please.

The APR -- yes. The APR1400 does not have an intervening non-safety bus, which is good, and to address the staff's concern regarding the common windings, the applicant committed to provide a failure modes effects analysis, or FMEA. The staff's technical concerns were in three areas, specifically, voltage regulation of the safety buses; transients caused by non-safety loads impacting safety buses; and failure points between the offsite power supply and the safety buses. Next slide, please.

Regarding voltage regulation, the staff found that the tap changers on the primary side of the transformers are adequate to regulate voltage in an acceptable range. Regarding transients failures on the non-safety side affecting the safety buses, the applicant's failure mode effects analysis showed that the safety systems would be able to perform their safety functions, and regarding faults between the offsite power supply and the safety buses, the applicant demonstrated to the staff's satisfaction that such faults would be detected and that the automatic bus transfer would transfer loads to the alternate power supply or to the diesel.

The staff finds that the applicant's response is adequate. We are planning to issue an RAI to close the loop on -- on the issue, but in the staff's mind, after reviewing the FMEA, the technical issues are resolved. Next slide, please.

The open phase issue is still an open item. The applicant provided their proposed design in November of 2016. The staff finds that the applicant's open phase detection system on the primary side is acceptable. However, they have not provided features for the safety buses that would protect equipment in the event of an open phase. There are a number of ways in which this issue can be resolved, and the staff has prepared an RAI to see which one they are going to choose.

And that is it. On to Chapter 10.

MEMBER REMPE: Before you do that --

MR. WUNDER: Whoops.

MEMBER REMPE: -- could I interrupt you with a question, please?

MR. WUNDER: I was hoping you wouldn't, but yes, sure.

(Laughter.)

MEMBER REMPE: Okay. During our subcommittee meeting, there was a lot of discussion about the APR1400 being a single-unit DCD, and in fact, in Section -- or Chapter 8, there were several responses to RAIs where the issue was closed because the applicant said hey, it is a single unit design. GDC 5 does not apply. And the staff responded, yes, you are right.

But then if I go to Section 8.3.2, it discusses that -- that GDC 5 may be applicable to a COL applicant that references the APR1400 design if its application includes multiple units. And this is the section in the Chapter 18 about sharing the structures, systems, and components. So I have been thinking about it a little bit, and what would happen if a COL applicant came in and said yes, I would like to put two units in, and I would like to share certain auxiliary systems? How would the staff interact with a certified design for a single unit, and when they come in with the multiple unit, would you ever be able to unwind it and say --

MR. WUNDER: Yes. What they do is they come in with a departure, and they have to analyze it.

MEMBER REMPE: Everything? So --

MR. WUNDER: Yes.

MEMBER REMPE: -- I mean --

MR. WUNDER: Every --

MEMBER REMPE: Safety analyses, PRA, it would be very --

MR. WUNDER: I don't want to get into the details, but --

MEMBER REMPE: Yes.

MR. WUNDER: -- I was the lead on South Texas, which was a single unit certification, and they came in with an application for -- for two more units, and it is dealt with at the COL phase.

MEMBER REMPE: And it can be unwound enough --

MR. WUNDER: It can be --

MEMBER REMPE: -- to figure out --

MR. WUNDER: -- yes.

MEMBER REMPE: -- how to fix the PRA and --

MR. WUNDER: Yes ma'am.

MEMBER REMPE: -- everything like that? Okay. Thank you.

MR. WUNDER: Okay. Now on to Chapter 10, Steam and Power Conversion System.

The staff safety evaluation looked at the turbine generator and rotor, the main and auxiliary steam systems, main and auxiliary feedwater systems, condensers, circulating water, and steam generator blowdown. With the exception of five unresolved issues, the staff has found that all applicable regulatory criteria have been met. There were I believe 19 open items at the end of Phase 2. Of the five that remain, four are associated with the turbine generator speed control and overspeed protection, and one is a request that the applicant provide an auxiliary feedwater system reliability analysis. Next slide, please.

Most of our Chapter 10 efforts following Phase 2 have been in the area of turbine generator overspeed protection. The DCD does not specify a turbine design because they wanted maximum flexibility for any potential COL applicant. The staff's concern was that the amount of information that was included in the DCD might not constitute an essentially complete design, so we asked for and we received a conceptual design for the overspeed protection system.

However, we still need some more detail in order to reach our required conclusions. We have been working with KHNP on this, and we are going to meet with them a little bit later this month to come to a final resolution on precisely what we need to have in the DCD to -- to reach the required conclusions and to bring this item to closure.

And I mentioned that the other remaining open item was the AFWS reliability analysis. We have requested the information in a recent RAI, and we are going to have a teleconference with them to iron out the details of exactly what it is we want a little bit later this month. And I am --

MEMBER BALLINGER: I have a question --

MR. WUNDER: Yes sir.

MEMBER BALLINGER: -- about that. So what you are saying is that KHNP has provided the staff with what amounts to a theoretical design, or a -- an exemplar design for overspeed protection system. So an applicant were to come -- if an applicant were to come in and they wanted to buy a turbine from vendor Y, which has a slightly different or whatever overspeed protection system, how does that -- how do they deal with that? How do you deal with that?

Because it seems like there is a lot of effort gone into producing this exemplar design for overspeed system, whereas it is not going to be the same, except fortuitously, when a -- an applicant comes in with a real turbine.

MR. WUNDER: Right. And when an applicant comes in with a real turbine at the COL stage, it will be evaluated at the COL stage. But I -- I am not sure I am --

MEMBER BALLINGER: There is --

MR. WUNDER: -- grasping the --

MEMBER BALLINGER: -- really --

MR. WUNDER: -- question.

MEMBER BALLINGER: -- no connection. We are just having an -- a turbine design that sort of satisfied a set of requirements, but when the -- when the applicant comes in with their design, you start from zero?

MR. WUNDER: Well --

MEMBER BALLINGER: Not being a turbine designer, I am probably, you know, exposing my ignorance, but it just seems like there is a disconnect.

MR. WUNDER: Well, you know, you are not really starting from zero. You are starting with a conceptual design.

MR. DIAS: Yes. Excuse me, this is Antonio Dias again, Plant Systems, NRO. KHNP did not provide a conceptual design. They were actually trying to use a COL item, you know. They were not trying to use the CDI conceptual design information --

MEMBER BALLINGER: Okay.

MR. DIAS: -- solution. They were proposing a COL item where, you know, they were basically saying this is what the applicant, the COL applicant will have to address, okay? So this is what they did.

We consulted with OGC, and again, this is not final, okay, but we consulted with OGC, and their interpretation of what the rule says, the rule says for -- there is a need for a complete application except for things that are basically site-specific, and they -- in the rule, the example is for instance the water intake structure, okay, or some waste, you know, building. Other than that, everything would be, you know, literally considered, what is the design that they need?

So for this reason, this is not resolved yet, okay? And George Wunder is correct. There will be a meeting, I think it is next week. We are going to try to -- or later on, we are going to try to resolve this, you know? But right now, following advice from our legal counsel, we don't think that that is an acceptable process in the application.

MEMBER STETKAR: I think that this is an example of, from my perspective, that gray area between being very specific in a design certification document. In my opinion, the design certification document should specify needs, requirements. In other words, turbine missile frequencies are established, and that a protection system should provide assurance that those frequencies are not exceeded. Details, very, very specific details of how you accomplish that, obviously will depend on the individual turbine vendors' details of the design.

Now historically, and I hate to bring this up, the NRC has certified designs, and I will not name the vendors, where exceedingly little information was provided about any details of things like reactor protection, safeguards actuation, really really important systems, and those were all left to DAC and ITAAC. So from my perspective, trying to specify too much at the design certification stage can be dangerous. I think that that design certification should specify what needs to be accomplished, and the individual design details once somebody buys a turbine should demonstrate how that is accomplished.

MEMBER BALLINGER: Yes, and to amplify on that, in the discussion in the subcommittee meeting, we had exhaustive conversations back and forth about very specific pieces of the overspeed system, and so it strikes me that at that level, if that gets enshrined in some way so that when the applicant comes in with a real turbine, the comparison starts getting made with the detail --

MEMBER STETKAR: Now, that -- that is true, Ron, but that being said, in this particular instance, the applicant, for whatever reason, decided to enshrine in the design certification document more details about certain aspects of the design than others have, and -- and that is -- that is their decision, you know? That may constrain a future turbine vendor in terms of taking departures from something in the certification document.

And here, again, you know, the specific details do not matter for the purposes of today's discussion, but indeed, they did specify some elements of the -- let me say the turbine protection design philosophy regarding electrical overprotection versus mechanical overprotection and what has higher priority and things like that. That is their decision in the design certification document, and a later vendor will have to deal with that. Trying to increase the specificity in the design certification document may not be consistent with what the staff has accepted in other elements of design certifications where details of -- of designs have been left to the -- once the combined license is issued.

MEMBER BALLINGER: So it is --

MEMBER STETKAR: So --

MEMBER BALLINGER: -- what -- what you asked for?

MEMBER STETKAR: It is -- in this case, the decision to put some details in the design certification document can lead the staff to ask for more and more and more details, which may not be consistent with what is done in other areas, where the decision is not to include very many details in the design certification document, where the staff will accept that and say yes, we will work out the details later.

MEMBER BROWN: I want to make one observation. Excuse me? My mic is on. Can you hear me?

Oh, I am sorry, I will wait.

Oh, okay. There was a lack of detail, but -- in this particular discussion. We went through a lot of it during the subcommittee meeting, and my -- my biggest concern when I -- was the ability to define the level of independence that was going to be specified. They didn't really talk to that. They talked about redundancies and things like that but were fundamentally not very crisp in terms of how independence was maintained.

You do have the separate mechanical

overspeed trip, so the argument I guess could always be met that your electronic systems and everything else that are in there also don't have to be -- doesn't matter what they are. It is -- if you're going to say that, you ought to say that. If you want it to be independent of the normal control, the overspeed stuff, then you ought to specify that. That is the level of detail I was looking for, some type of a one-line functional diagram that showed you did maintain independence between the normal control systems and the overspeed trip systems so that one could not compromise the other.

I only brought it up because a past project that I worked on, or at least consulted on, actually didn't maintain independence between the power supplies, and one power supply failed and took out the normal speed control, drove the thing to overspeed within about 30 to 40 seconds, and the overspeed trip function was disabled. And if it hadn't been for an operator who tripped the turbine speed, the trip valve before it -- literally oversped at about the 30 or 40 percent range, we would have had a serious problem of pieces of the turbine spinning all over a submarine. Not a friendly --

MEMBER STETKAR: Okay, Charlie. This was

not for a commercial nuclear power plant.

MEMBER BROWN: John, I understand that --

MEMBER STETKAR: No --

MEMBER BROWN: -- okay?

MEMBER STETKAR: -- but I just wanted to make sure that's on the --

MEMBER BROWN: Well let --

MEMBER STETKAR: -- record.

MEMBER BROWN: -- finish, okay? I am just -- my point being is that if you want it independent, you ought to say that. If you're going to accept it that it's not independent between your overspeed and your normal control, then you ought to say that and illustrate it as part of the DCD. That is my only point. I don't need, you know, excruciating detail, but you ought to at least have some specificity relative to the relationship between your electronic normal controls and your electronic overspeed controls.

MR. WUNDER: Thank you.

MEMBER BROWN: That -- that was my only point from the previous meeting. I just looked up my comments just to make sure I got it right.

MR. WUNDER: Thank you, and Dr. Ballinger, is your question answered satisfactorily?

MEMBER BALLINGER: Yes.

MR. WUNDER: I am out of slides, so I am going to turn it back over to Tarun Roy.

MR. ROY: Chapter 11, the Radioactive Waste Management: the SER for Chapter 11 addresses the source term, liquid waste management system, LWMS, gaseous waste management system, GWMS, solid waste management system, SWMS, and the process and effluent radiation monitoring and sampling system, PERMSS.

There are three open items that remain to be resolved, and they are under NRC evaluation right now, and that was discussed in the last subcommittee meeting in October. And these are the three items, actually. The first item, the seeking DCD updates for the liquid effluent tracking process for detergent rad waste system, and there are two questions on the request for additional information on the descriptions provided for the GWMS radiation monitoring and LWMS radiation monitoring, two questions. That's it.

MR. CIOCCO: That is everything for the staff, yes.

MEMBER BALLINGER: Okay. We are scheduled for a break at 9:50, but I need to appeal to a higher authority, namely Chairman Bley, about constraints on schedule because it is a Full Committee meeting. Do we need to adhere to the -- to the letter of this schedule?

CHAIRMAN BLEY: We have to adhere to the letter of this schedule. Your meeting runs from 8:35 to 11:30.

MEMBER BALLINGER: Oh, okay. So --

CHAIRMAN BLEY: If you want to have a break, you may do that.

MEMBER BALLINGER: Okay. So let's --

CHAIRMAN BLEY: I am sorry. Just a minute.

MEMBER STETKAR: What is the staccato between open and closed --

MEMBER BALLINGER: Okay.

MEMBER STETKAR: -- in the following --

MEMBER BALLINGER: That was -- that is -- that is --

MEMBER STETKAR: We need to --

MEMBER BALLINGER: -- the next step.

MEMBER STETKAR: -- hit any of the open --

CHAIRMAN BLEY: Oh, that is right --

MEMBER STETKAR: -- things --

CHAIRMAN BLEY: -- yes.

MEMBER STETKAR: -- on schedule.

MEMBER BALLINGER: The next group of

meetings is all closed.

MEMBER STETKAR: Everything until the end is --

MEMBER BALLINGER: No.

MEMBER STETKAR: -- all closed?

MEMBER BALLINGER: There will be a meeting on the KHNP presentations on the fluidic and CHF topicals are closed, and then the first part of the staff presentation is also closed, so it is -- it is a -- a continuous --

MEMBER STETKAR: So we come --

MEMBER BALLINGER: -- set.

MEMBER STETKAR: -- open --

MEMBER BALLINGER: At the very end.

MEMBER STETKAR: At the very end.

CHAIRMAN BLEY: Yes.

MEMBER STETKAR: Okay.

CHAIRMAN BLEY: Yes.

MEMBER STETKAR: I have no idea how that works, but --

MEMBER BALLINGER: In any case, we need to take a break so that we can set up for the closed -- closed session.

CHAIRMAN BLEY: So we will be closed, but then we will open again just for that last --

MEMBER BALLINGER: Yes.

CHAIRMAN BLEY: -- session.

MEMBER STETKAR: Do we -- do we need open -- do we need public comments at this point?

CHAIRMAN BLEY: I think we should do that, Ron. Why don't you --

MEMBER STETKAR: But we did in our last meeting.

CHAIRMAN BLEY: -- ask for public comments at this point. For everybody listening in, we will take a break, we will be going into closed session, and we are due to be open again at 10:45, is that correct?

MEMBER STETKAR: No.

CHAIRMAN BLEY: Well, split the difference between 10:45 and 11:25 because the first part will be closed.

MEMBER BROWN: The CHF part is open, and the fluidic device part is closed in that NRC presentation?

MEMBER BALLINGER: It is the -- the CHF is closed. The fluidic is open.

MEMBER BROWN: Oh. That is not what the schedule shows.

MEMBER BALLINGER: Yes, I know, it

doesn't. Things are fluid.

(Laughter.)

MEMBER BROWN: No, they are just wrong.

MEMBER BALLINGER: Things are fluid.

MEMBER REMPE: That's a nicer way of saying it.

MEMBER BROWN: They are just wrong.

CHAIRMAN BLEY: I guess that will work. At this point, take public comments on anything we have talked about.

MEMBER BALLINGER: Okay.

CHAIRMAN BLEY: And then before you declare a break, come back to me again.

MEMBER BALLINGER: Okay. So that being the case, I am assuming the lines are open anyway, but are there any public comments from the folks in the room?

(No audible response.)

MEMBER BALLINGER: Hearing none, are there any folks out, public -- out in the public?

PARTICIPANT: It's probably not open.

CHAIRMAN BLEY: It is open. Just ask for --

MEMBER BALLINGER: Okay. Are there any public comments?

(No audible response.)

MEMBER BALLINGER: I am not sure that it is getting out there.

CHAIRMAN BLEY: That's all we have. Okay.

MEMBER BALLINGER: Okay.

CHAIRMAN BLEY: Thank you for --

MEMBER BALLINGER: Thank you.

CHAIRMAN BLEY: -- for -- as I best understand this, sometime between 10:45 and 11:25, we will reopen the meeting. For anyone who wants to make a comment from the public on that last session, we will ask for public comments at 11:25 this morning --

MEMBER BALLINGER: Yes.

CHAIRMAN BLEY: -- once more. And at this point, we will --

MEMBER BALLINGER: Break for -- it is scheduled for 20 minutes, and that is probably going to be good enough.

CHAIRMAN BLEY: We will recess --

MEMBER BALLINGER: So that would be --

CHAIRMAN BLEY: -- until --

MEMBER BALLINGER: -- 5 after?

CHAIRMAN BLEY: -- 5 after.

MEMBER BALLINGER: It's closed. Come back at 10:05.

(Whereupon, the open session went off the record at 9:47 a.m. and resumed at 11:03 a.m.)

MEMBER BLEY: We're out of the closed session and we are into the open session. So you can keep going.

MR. LU: Oh, okay. So from - based on our initial interaction between perhaps the system and them, Chapter 3 reviews I think at this point is you are cracked if you have cavitation, and that especially the ECCS injection pump is either you have the cavitation, and the vibration is so large that's really the point is the big space will fail because of cavitation.

But the difference between this device and the pump is this nozzle is stationary. So when it's stationary under the symmetry nozzle - so when it does have cavitation but it may not have as much vibration as a ECCS injection pump, which is continuously spinning with that cavitation, that's going to be very bad.

So that's our initial talk and we were going to get back to you when Chapter 3 review will come back.

MEMBER SKILLMAN: We'll adjust it - we will address it then.

MR. LU: Yes.

MEMBER SKILLMAN: But just in a quick response, understand for the fleet that you've analyzed today the injection comes from either an RWST or a BWST where your maximum gas concentration is 8 PPM of atmospheric oxygen.

Here, you're running without an LPI, that low pressure injection, and you have 300 to 500 seconds, five minutes plus, with water that is fully saturated with dissolved nitrogen. Very different situation.

MR. LU: Yeah, we understand.

MEMBER SKILLMAN: So let's talk about this in Section 3.

MR. LU: Yeah, yeah, yeah. In Chapter 3 I think there.

MEMBER SKILLMAN: Thank you.

MEMBER BLEY: Ron, back to you. Everything's open. We are in open session and so continue on. Continue on.

MS. UMANA: Okay. I just want to confirm. Alexander, are you on the line?

MR. TSIRIGOTIS: Yes, I am.

MS. UMANA: Okay. We are about to start the presentation on the topical - on the fluidic device. So I wanted to confirm that you were there. Okay.

My name is Jessica Umana. I am the project manager for the topical report on the fluidic design that - fluidic device design, and Matt Thomas, who is sitting over yonder, is the technical reviewer who led the review on this.

So today, I am going to cover the areas of the fluidic device topical report that we reviewed. I am going to provide you with an overview of the safety evaluation and then briefly discuss some issues that came up I believe in - during the review and in the subcommittee.

Finally, I'll draw everything into a nice packaged conclusion for you.

So the areas of review - the staff areas of review covered the overall design concept and operation, the full scale test at the VAPER Test Facility, dissolved nitrogen effect and the uncertainty analysis.

In this evaluation, the staff approved the applicant's development of the safety injection tank fluidic device in conformance with a specific set of design and performance requirements of the APR 1400.

The staff also approved the full-scale

testing results to meet the applicant's specific set of design criteria. This safety evaluation did not provide an approval for the safety injection system design requirements which are intended by the applicant to comply with GDC-35 and 10 CFR 50.46. That will be looked at as part of the applicant's topical report.

These issues came up in the subcommittee - the effects of small break sizes on the safety injection site fluidic device performance.

Shortly after the subcommittee KHNP submitted thoughts of the half pressure sensitivity test which indicates that no, there is no break size that could result in the short circuiting of the safety injection tank covered gas and this seems to be somebody's - everybody's - I am sorry, everybody's favorite topic right now is the effects due to cavitation.

So this was brought up in the subcommittee and the staff has since engaged with KHNP to discuss this and it will be addressed, this part of Chapter 3 - of the Chapter 3 review.

Right now the staff is looking for information from KHNP. I think I'll probably - if you want to know specifics I can provide them for you.

But I'd rather let the technical staff let you know exactly what it is they are asking for.

We have assessment internal discussions going on within the staff to make sure that the methodology that's used to address this issue is reviewed correctly by the staff.

And finally, the conclusion - the full-scale test facility provides sufficient and adequate means for testing the safety injection tank fluidic device to validate the performance of the it against the APR 1400 design requirements.

The full-scale tests demonstrate and confirmed that the safety injection tank fluidic devices pass the flow of control.

The performance and design of the safety injection tank fluidic device tested in the VAPER facility satisfies the design requirements of the APR 1400 fluidic device.

Manufacturing tolerances and dissolved nitrogen have insignificant effect on observed pressure loss coefficient.

The design requirements of the APR 1400 bound all full-scale experimental results with uncertainties. Therefore, the topical report for the fluidic device was found acceptable by the staff.

So any questions? No?

MEMBER KIRCHNER: Just a minor question, Jessica. What does acceptable mean versus approved? You used the term on the - you approved the development of the -

MS. UMANA: I think I used them interchangeably, actually. So -

MEMBER KIRCHNER: Okay. I just wanted to understand if there was anything implied by acceptable -

MS. UMANA: No.

MEMBER KIRCHNER: - versus approved. Thank you.

MS. UMANA: No.

MEMBER MARCH-LEUBA: I believe approved - it becomes approved when it goes all the way to the top and the lawyers have signed it, right?

MS. UMANA: The what to top?

MEMBER MARCH-LEUBA: During your reviews you find all the approach acceptable.

MS. UMANA: Yes, on every -

MEMBER MARCH-LEUBA: And eventually, there is a review of your review and it gets signed by the lawyers upstairs and then it's approved. You don't have the authority to approve this?

MS. UMANA: No. No, I don't.

MEMBER MARCH-LEUBA: Right. So she determined that it is acceptable and then the lawyers approve it.

VICE CHAIRMAN CORRADINI: So that's because I think we are done, I sense. So is it Chapter 6 that we will see the - this design incorporated into a broader analysis of the LOCA methodology or is it in Chapter 15? I just want the make sure?

MR. LU: Chapter 15, where it will be relate it to large-scale LOCA topical.

VICE CHAIRMAN CORRADINI: Not in Chapter 6?

MR. LU: No.

MR. CHON: This is Woochong Chon from KEPCO Nuclear Fuel. There is one section, 6.2.1.5. There is one section.

VICE CHAIRMAN CORRADINI: Thank you, because I thought that's where you referenced this in the subcommittee meeting and I went looking for it. Okay.

MR. THOMAS: This is Matt Thomas.

VICE CHAIRMAN CORRADINI: So it's in both places.

MR. THOMAS: Yeah. This is Matt Thomas, NRC. It's correct, this topical report is referenced in Chapter 6.3 but the actual performance analysis is done in Chapter 15.

VICE CHAIRMAN CORRADINI: Okay. Thank you.

MS. UMANA: Okay.

MEMBER BALLINGER: Thank you.

MS. UMANA: Okay. Thank you.

VICE CHAIRMAN CORRADINI: Can I ask another question?

Okay. So just to get to Member Skillman's question about cavitation, which you will address in Chapter 3, this is driven by a different phenomenon than pumps far away.

This is driven by essentially, potentially, vapor formation and going away right at the DVI line into the down comer.

So where is that source term going to be computed or estimated or at least bounded by the robustness of the - of the mechanical component. You don't have to answer now but something to think about because I think where Member Skillman is coming from is reasonable. It's just we wanted to make sure the envelope - this falls within an envelope.

MEMBER SKILLMAN: Okay. We understand.

VICE CHAIRMAN CORRADINI: Okay.

MEMBER SKILLMAN: Thank you.

MEMBER BALLINGER: Okay. I think at this point we need to now take public comments again for this one section. Are there any people in the room that would like to make a comment?

Okay. I am assuming that the line is open, based on the cracking and popping and stuff. Is there anybody in the - any member of the public outside that would like to make a comment?

Hearing none, then I think that we need to go around the -

MEMBER BLEY: No.

MEMBER BALLINGER: No? Okay. In that case, I turn it over - turn it back to the chair.

MEMBER BLEY: Thank you, Ron.

This isn't a subcommittee meeting. We did tell the people earlier that we would take comments at 11:25. So we will do that but at - I may do something funny here.

At this point, we are going to go off the record until 11:25 so don't run away. And then I'll open it again then to see if there is anybody who wants to make a comment.

So we are off the public record for 10 minutes. We aren't recessed either.

(Whereupon, the above-entitled matter went off the record at 11:13 a.m. and resumed at 11:25 a.m.)

MEMBER BLEY: Please, could we have quiet? We have opened the phone line again. It's 11:25.

I just wanted to check and see if there is anyone on the phone line who has come back and would like to make a comment on the record.

We are ready at this time. Please go ahead. Going, going, gone. We are off the record again and we are recessed until 1:00 o'clock.

(Whereupon, the above-entitled matter went off the record at 11:25 a.m. and resumed at 1:00 p.m.)

MEMBER BLEY: The meeting will come to order at this time. I am going to turn it over to Pete Riccardella for the next discussion.

MEMBER RICCARDELLA: Okay. So we now move on, we are all excited about I am sure.

MEMBER BLEY: On the edge of our chair.

MEMBER RICCARDELLA: Sorry. Do I need to start over again?

MEMBER BLEY: We don't have - we don't have either Corradini or Stetkar here to enforce.

MEMBER RICCARDELLA: This afternoon we are going to be briefed by Rob Tregoning on Reg. Guide 1.207 and the associated backup documents in NUREG CR 6909.

I will point out that a prior author of 6909 Rev 0 was our own Dr. Shack. Rev. 0 of these documents were published in 2009 and we are now going to be reviewing a proposed revision to the two documents, Rev. 1 to both the Reg. Guide and the NUREG.

We have had subcommittee meetings on these in December 2014 and more recently in December 2016. Rob - and I'll point out that one of the authors of the - of this work is Gary Stevens who no longer works for the NRC but I believe he's on the telephone and will be able to participate if we need him.

Rob will be presenting the background on environmental fatigue, the current guidance and the proposed revision to the guidance.

We will now proceed with the meeting. I call on John Nakoski to make some introductory remarks.

MR. NAKOSKI: Thank you. I am John Nakoski. I am the acting deputy division director for the division of engineering and research.

I appreciate the opportunity to come and brief the committee on this topic. This is a longstanding research effort on behalf of the NRC since the 1990s.

Its focus is to support regulatory decision making for existing light water reactors and new reactor applications.

What we are discussing today is meant to consolidate all the prior guidance for addressing environmental effects on metal fatigue.

Further, Reg. Guide 1.207 is an integral part of the guidance on environmental assisted fatigue within the draft guidance document for subsequent license renewal.

As you're aware, the technical basis supporting Revision 1 of Reg. Guide 1.207 is NUREG CR 6909, Revision 1.

This document builds on the knowledge contained in the original version of the NUREG and strengthens it by considering almost twice the experimental data for applicable - sorry, ferritic and stainless steel and nickel chrome iron alloys, developing adjustment factors for air fatigue design curves, developing improved environmental correction factors expression, conducting analyses that validate the environmental correction factor method and providing a sample problem to provide additional user guidance.

We received substantial public comment and have carefully considered them in the development with appropriate responses to identify needed changes to both the Reg. Guide and NUREG reports.

We also want you to recognize that none of the comments that we have received has resulted in significant technical changes.

The approach has thus far been demonstrated to be appropriately conservative for regulatory action.

I am looking forward to ACRS feedback on this guidance and our plans are to finalize both the Reg. Guide and the NUREG in advance of issuing the subsequent license renewal guidance document this year.

And now I'll turn it over to Rob to give you the technical content of the presentation. Thank you.

MR. TREGONING: Okay. Thank you, John, and thank you, Dr. Riccardella, for your introduction and I wanted to thank the chairman and the members for allowing me to speak in front of you today.

I'll apologize to those that were here in December for the subcommittee meeting and I've already sat through this rather dry topic once already.

I regret to inform you that the slides are not changed substantially from that meeting. They are merely condensed to fit in the time period that's allotted so - and I apologize in advance for those that are sitting through this for the second time in - within the past two months.

So I am here to talk about proposed Revision 1's Regulatory Guide 1.207, which is on incorporation of environmentally-assisted fatigue effects, end-fatigue analyses of metal components.

Dr. Riccardella gave a good issue summary. I am going to maybe expand on what he summarized here a little bit. As he mentioned, we are revising the guidance for environmentally-assisted fatigue - the current guidance, Rev. 0 of 1.207.

We put out a draft guide, DT-1309, which I'll be discussing. That was the proposed revisions to this - to that original guidance - and then the supporting technical basis is Revision 1 to NUREG 6909.

Again, we are revising Rev. 0. So those will be the two documents that I am really talking the most about today.

As Dr. Riccardella mentioned, we briefed the ACRS Metallurgy and Reactor Fuel Subcommittees back in December of 2014, so a little over two years ago.

We released both these draft documents for public comment in the 2014 - late 2014, early 2015 time frame. That's why you see the span of years there.

We got a lot of comments, as you'll see. So it really took us quite some time to respond to the many comments that we got.

That's why you see about 18 months between 2015 and 2016 spent figuring out how to best address and then respond to the public comments.

And then at the same time in parallel we, of course, have been modifying both of these documents.

As Dr. Riccardella mentioned in - less than two months in December we briefed the Metallurgy and Reactor Fuel Subcommittee on the public comments and the changes to both documents and as John mentioned we are here today soliciting ACRS support for issuing the final regulatory guidance.

And one of the things you're going to hear next month in great detail about the subsequent license renewal guidance documents, right now this Rev. Guide and this revision has mentioned several places within those subsequent license renewal where I'll use the vernacular - the acronym SLR - so I apologize if I lapse into acronym speak from that perspective. But those SLR guidance documents actually reference these - this revised Reg. Guide.

So the plan has always been to release this or finalize this guidance in advance or in parallel to those SLR guidance documents.

So, hence, the schedule that you're seeing while we are coming and then early next month you're going to hear about the SLR guidance documents. And I'll show you a little bit throughout the presentation how this fits within those SLR documents.

So as mentioned, I am going to provide a brief background on what environmentally-assisted fatigue is, what the current NRC guidance is on environmentally-assisted fatigue, and then why we decided to revise that guidance.

And then I'll walk you through both the technical basis document, which is NUREG 6909, again from a high level and provide a summary of the public comments and a sample of a few responses of actually comments that we got and then talk about changes to the documents that we have enacted as a result of those comments.

And then I'll follow the same script for the Reg. Guide itself, provide an overview of the public comments, some sample public comments, changes to the documents. And then my last slide is on current status of these documents and planned next steps.

So, as I mentioned, the first piece of this presentation will be a background on environmentally-assisted fatigue.

So the first part of environmentally-assisted fatigue is this cumulative usage factor, which is a bit of a mouthful, or CUF we say for short.

This is, basically, how a designer would do a fatigue analysis. So he would look at his load history or stress history. He would look at his cycles.

He would have a certain number of cycles at each load history and basically using a linear or what's called in - or a minor's rule approach he would make sure that his number of cycles do not exceed the total allowable number of cycles that he can have for that specific component and that's government, quite simply, by a CUF factor of one. If the designer can demonstrate that a CUF is below one that component is good.

If he cannot demonstrate the CUF is below one then there is other measures than he might have to take to ensure or provide assurance that that component will perform its intended function.

I am not going to get into the different things he can do if his CUF analysis comes up greater than one. What we will be talking about here is just basically components of that CUF analysis.

So, again, as I mentioned, this little n in the equation that's the number of applied cycles for some loading sequence and then the big N is the number of allowable cycles that you can have for that particular load, and then Z is the number of different loading sequences that he applies so he just sums up all the individual contributions and has to show that they are less than one.

So this big N is a function of the alternating stress and the amount of load that you put on the component and it's also material dependent, and what I am showing on the right is actually the design curve for austenitic stainless steels.

So the designer basically has to

demonstrate that he is staying on the left part of this curve and not exceeding this curve.

These curves are provided in ASME Section 3 and, again, as I mentioned, they are for different materials.

Now, the most important point to keep in mind is these fatigue curves are based on empirical fits of air test fatigue data, so data in air, and then they also have been adjusted with design factors to account for aspects like data scatter, size effects, surface finish and atmosphere.

Notice I use the word design factor intentionally and not margins because it's not intended to be a margin per se. It's - these knockdown or design factors are to account for things that aren't simulated in the experiment but yet may be actually part of a realistic component, either design or environment.

So as I mentioned, CUF and the ASME design curves only consider air and this regulatory guidance is really considering the question well, what happens if have an environment other than air that your component is immersed in.

As I mentioned, those fatigue curves were developed from laboratory tests of small specimen -

polished specimens that were tested in air.

And the way those curves were developed they were best fit curves based on laboratory data and then those best fit curves were adjusted downward. Well, first they were adjusted for mean stress effects and then there were factors of two applied on the strain amplitude which, basically, knocks the curves downward and then 20 on cycles, which knocks the curves to the left.

So you see there on the chart the dash line is really the best fit curve and then the solid line is the design curve.

Now, what you find if you do testing of those same small-scale polish specimens but in a water environment - a high-temperature water environment instead of air - what you get happens are these red data points which land to the - you know, far to the left of those design curves, meaning that you have less life than you would be predicted to have if you use the design curves.

The blue symbols here are actually water tests but they are water tests that don't exceed any of the threshold levels that you would expect environmental effects.

So as you can see, those particular tests actually follow the air curve quite well. It's these other results which do not follow the air curve that we are concerned with here. And what you actually see we have many of those tests actually falling below the actual design curve for air.

So this regulatory guidance is designed to account for the effects of environment on your actual component.

And the way it works it's quite a simple approach. It just uses a simple environmental fatigue correction factors and it's defined, again, simply as a ratio of fatigue life in air at room temperature to the fatigue life in water and, again, these Fen factors are specific to the specific transient that you're considering.

So just like the CUF analysis you have to count up all your individual transients and add them together. With the environmental fatigue correction factor approach you have to develop Fen factors for each of your transients, apply them to your cumulative usage factor for those transients and then sum them up.

And at the end of the day, you end up with this CUF for environmental effects which we deemed CUFen. So it's exactly analogous to the CUF factor that you would get in air but it just incorporates the effects of the environment.

And I've shown here in this slide an equation for Fen that's in the NUREG for stainless steel materials just to give you a sense for what some of the things that are modeled.

At least for stainless steels, there is explicit consideration of temperature, the amount of oxygen in the environment and then the transformed strain rate.

Again, this is the - actually this is the expression that's in Rev. 0 so the prior Fen expression.

MEMBER SKILLMAN: Rob, let me ask this. Just looking at your first bullet, Fen is the - in an air versus - or over N water, is that number always less than one?

MR. TREGONING: It should always be greater than one. Greater than or equal to one. You should always get greater life in air than in water or at least equivalent life.

MEMBER RICCARDELLA: Now, this is from Rev. 0. For those of you who are mathematically inclined I'll point out that the exponential of .734 is about two.

So even when those - all those environmental factors reduce to zero, in the old version in Rev. 0 you still got an Fen of two, at least for the stainless steel.

I don't know - I haven't looked at the other equations. So that's one of the reasons that they are revising - that a revision is in order.

MEMBER SKILLMAN: Thank you. That's helpful. Thank you.

MR. TREGONING: Dr. Riccardella, that was mentioned. Several people came up to us after Rev. 0 was published and said what are you guys doing - you're, clearly, idiotic because there can't be an Fen factor greater than one if you don't have any environment.

And if it's simply - to be honest, if it was simply a function of the fact that, you know, the curve fits that were done there weren't constraints put on them to make sure that they - you know, that they hit the right bounding value.

So that was something that we were - we were very keen to fix in Rev. 1 and we have certainly done that.

Okay. So the next thing I want to touch on is the current NRC guidance on EAF and as part of this I am going to give you sort of a sneak peek or a prelude of how that guidance is evolving, at least for subsequent license renewal and, again, you may or may not hear about this more next month because, again, it's a detail among the grand scope of everything that's happening in SLR space.

So as Dr. Riccardella mentioned, we have been doing - NRC has been sponsoring research with respect to environmentally-assisted fatigue for over 20 years now and some of the earliest work was carried out by - at A&L by - with Dr. Chopra and Shack.

They had separate NUREGs in the late 90s, one on low alloy steels and one on austenitic stainless steels documenting environmental effects.

And those NUREGs actually formed the basis of the earliest guidance that we gave in this subject on the GALL report in 2001 and specifically if you look at Chapter 10.M1 there is guidance in there that references these particular NUREGs.

Again, as Dr. Riccardella also mentioned, in 2007 we consolidated and updated the guidance that we had at that time in NUREG CR-6909 and those - and that's the current guidance that exists today.

So what do - we actually have two spaces of regulatory use of this guidance. We have the -

actually, three. We have the original licensing period, the license renewal period and then new reactors.

Plants in the original license period they have no guidance or requirements for considering EAF at all. So they have to do their CUF analyses but they don't have to consider environmental effects.

License renewal period, especially recent applicants, tend - I mean, they can take exception to this but the guidance that we have out there basically instructs them to use through NUREG 1801 Rev. 2 to consider EAF effects and we give them a variety of ways to do this.

They can either use the old NUREG, NUREG 6909 or an NRC-approved alternative. That's the same with carbon and stainless, and with nickel alloys we didn't - we didn't have a NUREG before 6909 so their option is to use 6909 or an NRC-approved alternative.

At least for subsequent license renewal the current document that's out there that's proposed for finalizing the language states that an applicant may either use 6909 Rev. 0 as long as they use a correct average temperature for each of their transients or they can use Rev. 1 of 6909 or, as is always the case, they can propose an alternative that the staff decides, then reviews and accepts as necessary, assuming that we assume it's acceptable.

Now, new reactors are in a little bit different situation. The original version of Reg. Guide 1207 was specifically only for new reactors.

In fact, we said it in the title. At the end you see four new reactors. So this - when the Reg. Guide was initially put out we limited the applicability in 2007 only to new reactors.

The technical basis for that original version of the Reg. Guide is Rev. 0 of 6909 and, again, consistent with what was done in Rev. 0, Revision 1 of 1207 basically just adopts the Fen approach summarized in Appendix A of NUREG 6909.

The original version of Rev. - of 1207 uses Rev. 0 Appendix A and then the proposed Revision 1 of 1207 references Appendix A of Revision 1 of NUREG 6909.

Background - let me finish with the background portion, talking about a little bit more in detail of the proposed revision for the guidance, why we decided that we wanted to come out with a Revision 1.

The first rationale, we wanted to consolidate existing guidance and also account for new information. There has been a lot of additional data that we have been - that we have been able to access since 2007 and we also wanted to update the guidance based on stakeholder feedback.

Dr. Riccardella mentioned the common one, the fact that the Fen factors didn't approach values of one when threshold limits weren't exceeded. That was, clearly, something that we had to change.

With respect to 1207, significant changes. The Reg. Guide was made applicable to all light water reactors. So we took the words, or new reactors, explicitly out of the title. It was probably one of the biggest noticeable changes.

The other thing with respect to applicability, and this is Rev. - this is the draft that went out for public comment.

You'll see that we have changed this a little bit in the final. But at least in what went out there we said it was applicable to components that are exposed to LWR environments that have a CUF calculation as required by the plant's current licensing basis where CLB - and I'll use the term CLB or the acronym CLB elsewhere in this talk.

We also revised the background section a bit and then the other big change, of course is that we revise the Fen equations as has been - excuse me - documented in Revision 1 of NUREG CR-6909.

MEMBER RICCARDELLA: So just to expand on that point two there, the Rev. 0 of the NUREG was limited to pressure - class-one pressure boundary components, right?

MR. TREGONING: I don't - I don't know that we explicitly said -- I don't -- we don't have applicability statements within the NUREG itself. Rev. 0 of 1.207 -

MEMBER RICCARDELLA: Yeah, that's what I meant.

MR. TREGONING: -- set pressure boundary components.

MEMBER RICCARDELLA: Yeah, that's what I -- I was referring to the --

MR. TREGONING: Oh, I am sorry.

MEMBER RICCARDELLA: -- to the Reg. Guide, not the NUREG.

MR. TREGONING: I either misheard you or --

MEMBER RICCARDELLA: Or I might have misspoken.

MR. TREGONING: There is too many Revs and numbers to keep in my mind. So if I -- if I misspeak between Rev. 0's and Rev. 1's and Reg. Guide 1.207 and NUREG 6909 I apologize. But yes, Dr. Riccardella's right.

MEMBER RICCARDELLA: But that -- that is somewhat of an expansion of applicability from -

MR. TREGONING: Yeah. That -- that's -- that's --

MEMBER RICCARDELLA: -- pressure boundary components to all components that have a usage factor calculation.

MR. TREGONING: And I'll touch on that a little bit more in detail as we -- as we go through this presentation.

So a little bit -- I want to give you a little bit about what changed in the draft NUREG itself and what didn't change.

So with respect to the air fatigue curves, there was no change at all. So if you look at Rev. 0 or Rev. 1, the air fatigue curves, the mean data curves are exactly the same.

Also, the adjustment factors that were used for Rev. 0 and Rev. 1, specifically the adjustment factor of 12 on cycles that was retained.

But that was one of the -- we presented our evaluation and rationale for this in the NUREG and this was one topic that we specifically requested public feedback on when we sent the NUREG out for public comment.

The air design curves are the same. The only slight difference is we were -- we were more clear in Rev. 1 to recommend for nickel-chrome-iron alloys the use of the stainless steel design curve.

And this is conservative -- most nickel-chrome-iron alloys actually have better fatigue life than stainless steel but we thought -- but the flip side is there is just not as much data for nickel-chrome-iron alloy so we felt like that was a reasonable position to take.

Now, with respect to the air curves, I just mentioned that there was virtually no change between Rev. 0 and Rev. 1. Now, what about with respect to the Fen changes?

So if you look at Rev. 0, we -- Rev. 0 has different Fen expressions for carbon and low alloy steel as we have already talked about extensively the Fen could be greater than one even if you didn't have environmental effects, and this was clearly in error.

And we also had different expressions for the stainless and the nickel-chrome-iron alloys. So what have we done in Rev. 1?

Well, we now have just the single unified expression for all ferritic steels -- carbon and low-alloy steels. Fen is now one for all those alloys if there are no environmental effects.

The stainless and the nickel-chrome alloy Fen equations have a similar functional form -- in fact, the same functional form, just different constant values.

And I showed some plots here and I apologize, they are not -- they are not that legible. But the solid lines are the new expressions and the dash lines are the Rev. 0 expressions for Fen.

And then there is a chain dash line that actually represents what's done in the Japanese code. And both of these plots are for carbon and low-alloyed steels. One of these is the Fen factor is a factor of strain rate. The other is Fen versus temperature. And you'll see that for most of the space the new expressions are generally less conservative. That's not always the case. There is certain conditions, especially if you look at low-dissolve oxygen where the newer expressions are a bit more conservative. But I think that general statement that the Rev. 1 expressions are, you know, for most combinations of variables they are generally less conservative.

MEMBER RICCARDELLA: Some of those curves go up to pretty high numbers -

MR. TREGONING: Yeah.

MEMBER RICCARDELLA: -- 20, 30, 40.

MR. TREGONING: Yeah.

MEMBER RICCARDELLA: Are those relatively unusual circumstances, Rob? I mean, are you --

MR. TREGONING: Yeah. So that's a good point and I did -- I don't know that I brought this up before. So that would be highly unusual.

I don't think we have seen an Fen analysis with an Fen factor greater than probably about six. And the other thing that we have seen, quite -- when you do the analyses, those transients that have high Fen factors you generally don't accumulate a lot of strain as a result of those transients.

So the end result is you don't take much of a hit on your actual fatigue life. So the strain range themselves associated with, like, very low -- think about it, because to get high Fens you need a very low strain rate, right.

MEMBER RICCARDELLA: Okay.

MR. TREGONING: So you tend not to accumulate much strain on your component over those low strain rates. So there is somewhat of a compensating effect that even though you have high Fen you have low applied strain so you don't take a big usage factor hit.

So yeah, these plots probably exaggerate sort of the scope and applicability of these Fen factors somewhat and I would think most analyses you're going to have Fen factors that are somewhere between, again, one and probably four or five.

And again, someone, like Gary Stevens -- I don't know -- I am not asking to try them but someone who does fatigue analyses for a living would be able to comment a little bit more eloquently than I could on that particular point.

MEMBER RICCARDELLA: Sounds like Gary can chime in.

MR. TREGONING: Gary, you want to chime in? I don't know.

MEMBER RICCARDELLA: Gary?

MR. STEVENS: I couldn't resist, Rob. I'll chime in.

So yeah, there is an example of this in the Appendix C sample problem and I think that got brought up as a question by the staff even.

And if you look in there you'll see through some of the detailed analyses that there are some quite high Fens. But as Rob indicated, it -- you know, they are applied to stress cases that don't have any significant usage. So they don't manifest out as a significant contributor and that's consistent with all observations we have ever seen for the higher multipliers.

MR. TREGONING: Thank you, Gary. I figured you would feel passionate enough about this that you'd want to weigh in. So appreciate you doing that.

So other significant additions to Rev. 1, and Gary, this is a great lead-in for this slide, actually. We did two things. We did an extensive amount of verification calculations performed on both specimen and component test data. There is six total test series that were analyzed.

We analyzed the methods using ASME code with the Fen factors and compared them to the predictions to what was measured experimentally. As long as the tests were on specimens the approach was generally pretty good.

We got fatigue lodged within a factor of two. For fatigue evaluation if you can get within a factor of two that's generally considered to be very good -- very good agreement.

The component test we didn't get as good agreement but what we found was that those component tests either agreed with or were conservative -- conservatively predicted compared to the experimental results, and there are some reasons for that that we go into in a little bit greater detail in the NUREG when we look at analyzing those tests.

And then the other thing we did in Rev. 1 that Gary alluded to is we put in a sample problem and we thought that this was important to do to give people a solved problem, that they could work through the Fen methodology and demonstrate that at least for the sample problem they are -- they are applying that methodology consistently.

And the sample problem itself we got a lot of positive feedback from stakeholders that liked the fact that we included that. In fact, in the public comment period we got several people that worked the example problem from soup to nuts and we got a lot of valuable feedback from the work that they did that allowed Gary in particular to go back and revise some of the specifics of that sample problem to make it more -- to make it less ambiguous to solve.

There were some -- there were some ambiguities in there that could lead to different results that we've hopefully cleaned up as a result of the public comment.

So that's it for the background. Now I want to go into the NUREG itself and provide an overview of the public comments.

So as I mentioned, NUREG 6909 Rev. 1 was sent for public comment in April 2014 and while we were certainly looking for public comment in all areas we specifically asked for feedback in the following three areas.

The first area was the extension of the best fit mean air curves for ferritic steels and by extension the prior curves ended at, like, fatigue lives of 10 to the sixth.

We were looking at extending out to 10 to the 11th, which, again, is pretty big extension. We wanted specific comments on those adjustment factors for, again, knocking down the mean best fit curve into a design curve.

So we wanted specific comment on how we had developed adjustment factors within the NUREG.

And then the third thing, which on face seems like a good thing, but was probably a little bit too ambitious, we asked for an accuracy check of all the technical content of the NUREG, particularly with respect to all of the numerical content of the report.

So we got a couple of glib comments back from the public that basically said hey, you gave us two months -- there is no way in two months we are going to be able to look at all the technical content of the NUREG.

So I think I recognize now that this was probably a little bit ambitious to ask of the public to help us out here. But I will say, again, with respect to the sample problem and things like that we did get a lot of very good substantive feedback where commenters went and looked at a lot of the technical details associated with the report.

Public comment period ended in June of 2014. So this next slide shows that we got formal public comments from 10 commenters and this is the table that we used to track those.

We also got three additional commenters that provided feedback after the public comment period officially ended. So they are not formally shown here but we, obviously, took their comments and we have addressed those comments as well. So we just incorporated them with all the other comments that we got.

So the way we did this, you know, it's

sort of process but feel like it's worth mentioning because it's how we structure our response document. We identified each issue that was raised by a commented and, again, we partitioned them first into single issues.

So what you see if you read the comments from many -- many of these individual commenters would have about 10 different ideas in one single comment. So they were very complex comments.

We broke those things out into single issues as best we could. So we call them subcomments here so they are -- they are part of the initial comment -- the more expansive comment that was provided by the commented.

So we have tracked these as, you know, using the following system where X, Y and Z -- X is the letter number that's -- we have assigned them.

I am sorry, X is the abbreviation that you see in the table on the far right column. Y is the letter number -- I am sort of cycling back and forth between slides 18 and 19.

But abbreviation in the letter number are actually redundant but, you know, we are nuclear engineers so I guess we believe in defense in depth. And Z is the sequential comment number and then A through Z would be the subcomment depiction.

So I mentioned we got a voluminous number of comments. If you broke these out, we got 254 unique subcomments or issues. Two hundred and thirty-five of those were from the 10 formal commenters and of the three additional commenters we only had five additional.

And then there were 14 other comments that came up either from the authors at staff that we -- as we were reviewing the document many times in part to the public comments we got we stumbled upon other things that we said oh yeah, we really need to fix this as well. So there were 14 other things that we tracked.

I think I mentioned this but the comments that we got back on the NUREG were really good. They were generally very highly technical in nature. They were thoughtful and, as I mentioned, they were often expansive.

This caused us a lot of trouble to try to deconvolve some of the comments but and I think we probably spent as much time trying to make sure we understood the points that they were making as we did in actually figuring out how we were going to respond to the point.

Most of the technical comments you could group them in the following sort of broad areas. There were many based on the scope Fen method -- what's this good for, how can you use it, when does it apply, when doesn't it apply. We got a lot of comments on the adjustment factor analysis and application.

Of course, that was an area that we asked about so we were happy to see that. We got a large amount of comments that were basically related to clarification -- hey, what are you saying here, or you said this, did you really mean that.

And, you know, as authors those kinds of things are valuable because when you write things you have the certain intent and sometimes that intent is not always conveyed in what you've actually written. So I think these clarification statements have hopeful allowed us -- the document to be much more readily and more clearly understandable by the stakeholders.

We got a lot of comments with respect to the relevance of this approach to nuclear plant applications. Those were comments along the lines of this is way too conservative and here's why, you know, because in actual plan applications you have this effect and this effect that you don't have in your laboratory specimens.

So we got a fair number of comments related to that and then we got a good number of comments that vented about ASME in general and the overall conservatism of ASME requirements in conjunction with the Fen.

Now, this particular NUREG doesn't go into -- didn't look at ASME requirements at all. So while we recognize and agree with some of the comments that were made, that was not the purpose of this particular NUREG. This particular NUREG was just to look at incorporating environmental effect.

I think in general, you know -- and then I did the accounting. We agreed with the vast majority of not only the actual total comments but also the individual subcomments, and I think I estimated over 95 percent. So, again, that just sort of -- just sort of echoes the fact that we got really good meaty technical comments.

I would say the areas that we disagreed with the comments were generally not technically significant with respect to the Fen method and I've listed some of the reasons -- some of the areas that we disagreed.

We had definitely disagreements on

interpretations of ASME code requirements. But that's a non sequitur. It doesn't relate to this NUREG at all. So even though we disagreed it's not really pertinent.

We had some disagreement on how -- on whether load sequencing effects should be applied or not. A raging controversy in this particular area is whether you should consider strain threshold or not. So we had a lot of philosophic -- or we had some philosophical comments on that particular.

We had a lot of -- we had a good amount of dialogue on the interpretation of the AREVA test results. These were one of the component tests that we analyzed and we orally got comments from the testers themselves so that's always helpful, and we got comments that we disagreed with on the high cycle cutoff of the design curve and this was another area that we asked about.

So next, I am just going to provide a couple of samples and, again, we went more -- in greater detail in sample public comments and responses in the subcommittee.

So I've only just kept a few of these to give you -- to give the main committee a flavor of some of the comments that we got and I've tried to provide ones in areas that we asked for.

So this first one is -- talks about the adjustment factors, and even though this is a subcomment there is actually two points that are articulated in this -- in this one sentence.

So and this was similar to a lot of the comments. They were very difficult to deconvolve. So the first point that the commenter makes is that all the reduction factors are considered independent. So that's point number one.

And then he goes on to say it's not accepted in all international approaches that a constant Fen independently of the number of cycles is not justified clearly. So that second part of that phrase is actually another point.

So the response -- we agreed with the commented that yes, we do consider the adjustment factors to be independent. We don't consider synergistic or correlative effects.

They could be there. We -- they would need a lot more study to tease them out in any level of rigor that we could quantify, and that's basically what we said -- that there is insufficient data developed correlation factors for more rigorous analysis.

And we do -- we went in and we clarified that point in the NUREG to make that as clear as we could.

We also agreed with the commenter's second report that the method does assume that Fen is a function of applied strain and we have clarified that in the NUREG. That's done for simplicity as much as anything. If Fen was variable with strain it would make it a much --

MEMBER RICCARDELLA: Not with strain. With cycles.

MR. TREGONING: What -- but cycles or strain are really independent in some sense, right? He's talking about -- he's talking about strain more than he's -- he's linking cycles to strain.

MEMBER RICCARDELLA: I see.

MR. TREGONING: Because we have considered it a separable approach. If it was a function of strain it would be a much more complicated analysis that would have to occur.

Yeah, and some of this, you know -- Pete, some of this is -- you know, this was Claude Fadey comment. So those of you that know Claude -

MEMBER RICCARDELLA: Yeah, I know Claude.

MR. TREGONING: -- know that sometimes his comments are difficult to interpret. So you're probably looking at that and saying well, how did you get that interpretation.

So there was some -- there was a little bit of clarification that was provided separate from the comments.

This next comment that I am going to touch on came from MHI and this was with respect -- the extension of the best fit mean curve in air from 10 to the sixth to 10 to the 11th cycles.

The talk was simply that hey, it's too conservative. So our response we agree with the commented that, you know, it's a design curve -- it's meant to be conservative and that we also said that what's in the Rev. 1 is identical to what's been proposed by -- within the ASME code and that the extension is based on data that has a large mean stress component and that if you have data or an application that you don't have such a large mean stress component.

Yes, this curve could be significantly conservative and we tried to make sure that that point was clear within the NUREG so that someone using this would recognize that fact.

But, again, we are basing the curve on

data. We are intending it for to be conservative as well as apply to a broad range of engineering components and oftentimes these do have -- need high mean stress loading components.

So that's why we think the current code proposals and what we have used within the NUREG is actually quite appropriate.

MEMBER RICCARDELLA: I am trying to understand it. If I got back to your slide 13 --

MR. TREGONING: Sure. Yeah.

MEMBER RICCARDELLA: -- how could -- how could you be less -- I mean, you're going horizontal from 10 to the seventh all the way up to 10 to the 11th. It'd be more conservative would be to be decreasing that as you go out, wouldn't it?

MR. TREGONING: Well, okay. But you're looking at from 10 to the sixth to 10 to the 11th. You could break that knee a little bit sooner back at 10 to the sixth to give you a higher fatigue life out of 10 to the 11th. So or --

MEMBER RICCARDELLA: Okay.

MR. TREGONING: You know, and some have argued do you really -- does it really flat line or does it continue to do gown.

MEMBER RICCARDELLA: Yeah. That would

mean the not conservative analysis.

MR. TREGONING: Right. Right. And that's what you're trying to weigh because, to be honest -- and the commenters brought it up -- you don't have a lot of data independent. That's a lot of -- that's a lot of cycles.

MEMBER RICCARDELLA: That's a lot, yeah.

MR. TREGONING: Well, yeah. So those tests are done in very high cycle vibratory fatigue using very small specimens typically just to accumulate the number of cycles. So yeah, you have to be very careful in how you interpret that data, to say the least.

Okay. Any other follow-on questions?

MEMBER RICCARDELLA: Twenty-seven million hours. One hertz would be 28 million hours.

MR. TREGONING: Yeah. They are done way faster than a hertz.

MR. STEVENS: This is Gary. I have a comment. The commented might also have been referring to the ferritic curve which is not as flat.

MEMBER RICCARDELLA: Okay.

MR. TREGONING: Yeah. Thank -- thank you for clarifying that here.

Okay. Any other questions before I go

into the next part of the presentation, which talks about the high-level changes or a summary of the changes to NUREG 6909 Revision 1?

Okay. So these are the modifications that we have made after the public comment and you'll see if you read the draft version and read the newer version that we have made significant modifications to the NUREG in an attempt to address virtually all the public comments.

The main things that we tried to do we tried to explain more clearly and completely all the technical bases and assumptions supporting the work. But, again, I think -- and we wanted to do this because oh, we wanted to summarize the current state of knowledge that exists in this area but also provide a foundation for continued research.

So this is not -- this NUREG is not meant to be the last word on this, just the current thinking and the hope is it'll serve a base -- as a basis for not only continued research but then also continued changes in evolution in the ASME code requirements as well.

So the NUREG has expanded significantly over time. Rev. 0 was a lean 120 pages and much like the waistline of most aging Americans it's gone up exponentially almost. Rev. 1 draft is up to 320 pages. And then between the draft and final we are now up to about 500 pages almost.

So where did we add that between the draft and the final? Not much in the main body. We only added about 10 pages to the main body. The main thing we added was this Appendix D, which is a compendium of figures. That was about 135 pages.

And this new Appendix E, which provides a comparison of equations in the different revisions of the NUREGs, that was about 12 pages. So I'll talk a little bit more about that in a minute.

MEMBER SUNSERI: Ron, just a -- just a quick question.

MR. TREGONING: Sure.

MEMBER SUNSERI: I mean, by incorporating all this information into the document are you locking yourself in to having to revise this as those figures change or what's the reference on the -- am I making myself clear? I mean --

MR. TREGONING: Yeah.

MEMBER SUNSERI: You could kick out to a document that's maintaining those things independently in real time and all that stuff or you can import them in and institutionalize them. I mean, are you -- are you building a trap?

MR. TREGONING: I think -- you know, I would envision -- if you look at our history we have revised the technical basis about every 10 years in this area.

I would think in another 10 years we will probably have to -- we will probably need to do something. As far as the figures themselves, I mean, you know, as we get new data -- as we get new information we can regenerate the figures pretty easily.

One of the things that Gary did, Gary set up a really good database as part of his work here and we have got all of this information sort of within the database as well as a compendium of references available electronically. So that's made it so, you know, with that database we can regenerate these things relatively easily. So the figures themselves aren't that big a deal.

What's always -- what always takes longer is if we have to go back into the text and figure out different aspects that we may need to revise over time. So I wish there were an easier way to do that. I mean, we have got everything electronically so we will make it as painless as possible when and if that time happens.

But, you know, whenever we revise NUREGs there is certainly a fair bit of overhead that's associated with that. So it's a bit of a painful process by design.

MEMBER SUNSERI: Yeah, and that's all I was asking. I mean, you know, knowing how the process is I didn't know if you were locking yourself into a specific slot as far as the technical information is concerned. But if you're comfortable with it -- if it's every 10 years then that's fine.

MR. TREGONING: Yeah, I think -- I think that's going to be about right and, again, there is things that we have done in the interim and things we can do in the interim without actually going through a full-scale revision of the guidance that we could do in the interim. And even if we needed to have a new regulatory position there would be interim ways that we could go about that without having to revise the regulatory guide again. So but that's a very good question.

Okay. So now this side just summarizes the major changes to the documents. I mentioned that we added about 10 pages to the main body. Most if not all of those 10 pages are in this new Section 1.5, which is right up front and covers the bases and assumptions of the methods.

So this was a lot of we wanted to make sure the scope, the assumptions, the applicability, all of these things that we -- that we had assumed implicitly were clearly stated within the NUREG.

So we tried to be very intentional about -- itemizing all these different assumptions and limitations so that, again, it could serve as a foundation for future changes in -- that would be promulgated in this area.

Another thing we did is this modified rate approach we moved it. It was part of the caulking in low alloy steel section when it was really meant to be applicable to all the alloys.

So we moved it to its own section, clarified the write-up. I mentioned this before that we reworked and revised the example problem in Appendix C. If you look at the draft Rev. 1 we had Rev. 0 equations and Rev. 1 equations and this confused at least a few of the -- a few of the public commenters.

And we agreed, yeah, it was probably not a good idea to have two sets of equations in one document. So if we just kept the new equations in the main body, what we decided to do is we added this Appendix E that all it does is list the equations in Rev. 0 and those with Rev. 1, and then it provides a mapping as well where it says equation A1 in Rev. 0 maps to equation E3 in Rev. 1.

So we provided that mapping as well. So, hopefully, this will be helpful for people that are -- that are using older versions or that the prior versions of Rev. 0 if they want to convert to Rev. 1.

I've talked about the figures. So we got several comments about yeah, these figures are really small and they are hard to see. So we -- the main body we used high-resolution images to replace the figures.

Now, while that's nice the thing that it does is -- so the NUREG has not only grown in terms of physical paper size but the file space has grown as well.

So the actual NUREG itself is 90 megs now with these new higher resolution images. So it's a little bit ponderous to work with but we thought, you know, in the effort of having these better figures that it was worth it.

The other thing that we did was we added a whole new Appendix D, which is a compendium of figures separate and these are actually in large high-resolution images of every figure that's in the main body. So if someone wants to get in and look at individual data points they can go to Appendix D and pick these things out.

MEMBER SKILLMAN: Rob, are those figures copies from other legacy documents or are those figures unique only to this document?

The reason I ask is if they are part of another document then you have change control and configurations.

MR. TREGONING: Yes. So we -- you know, all the figures are newly generated for this version, right. If you go into Rev. 0 a lot of these figures might look exactly the same but they've been regenerated. A lot of them have new data that's associated with them.

So everything is a new figure from that perspective. But, again, if you looked at Rev. 0 there are many similar figures in Rev. 0 as are in Rev. 1. So does that -- does that answer your question?

MEMBER SKILLMAN: But -- no.

MR. TREGONING: No?

MEMBER SKILLMAN: For the new Rev. are

there figures newly constructed with more detail or more high resolution that are left out from an ASME code or left out from some other -- from some other industry document?

MR. TREGONING: I -- I am thinking here to make sure I don't answer incorrectly. I don't think so. We certainly -- we certainly reconstitute the ASME design curve, right. But, you know, that's done -- you know, you lift out the equations and recreate the curve.

I don't think that we have any figure that's a lift out from any other document. Gary might want to correct me if I am wrong. But I don't think there is any such figure.

And, again, we have got 103 figures so I am sort of rolodexing in my mind each figure to see if there is any like that and I really don't think so.

MR. STEVENS: This is Gary. Rob, in Section 6 -- the validation -- some of those figures are lifted out of papers that we did the validation for. But with respect to the rest of the sections of the NUREG you're correct, all the figures were generated new for this report although some of them do contain, as you -- like you stated, the ASME code curve, which is locked in or whatever. But the figures were generated new for this report.

MR. TREGONING: Yeah. No, I am glad you brought up -- thanks, Gary -- I am glad you brought up Section 6. But I think in Section 6 those are figures like here's the experimental set-up that was used for this test. Those were the things that were lifted out of the papers, correct?

MR. STEVENS: Yeah, that's right.

MR. TREGONING: Yeah. So our analyses of those particular tests, those figures are all new. So yeah, thanks, Gary, for clarifying that because I hadn't -- those figures had slipped my mind.

So does that answer your question now or --

MEMBER SKILLMAN: Yeah, it does, but it raises another question and that is if I am a -- I am an analyst deep in this technology am I know driven to compare CR 6909 Rev. 1 against some prior version of the ASME code and now I am like a man who has two watches -- what time is it.

MR. TREGONING: Well, again, 6909 Rev. 1 is distinct from the ASME code and if the Fen approach is -- again, it's a distinct approach. Now, when we have compared -- when we have use the code and Fen together for comparison it's done exactly the same way an analyst would do it.

So, you know, any analyst that was doing a fatigue evaluation they would go to code first and that would be the first thing they would do to construct their fatigue analysis and then they would go to 6909 Rev. 1 and do the Fen component of it, right.

So we are not superseding ASME code requirements in any way, if that's the concern. We are just adding a -- we are almost like adding a module to that evaluation to account for environmental effects.

So you'd go through your standard ASME analysis first, then you come to this module that's summarized in 6909 and figure out how you need to account for Fen effects on top of that analysis.

MEMBER SKILLMAN: Thank you.

MR. TREGONING: Okay.

MR. STEVENS: So we should probably clarify one thing on that, that the updated guidance requires you to use the -- I want to say the updated design curves in the report for stainless, which are equivalent to the code from 2009 to current and then you have an option of using either the design curves for the ferritic materials or the code curves -- recent code curves, which are more conservative.

MR. TREGONING: Right. Yeah. Thanks for clarifying, Gary. So we -- the stainless curves are the same. Like you said, we provide an option but it's clarified in NUREG that either option is appropriate.

Okay. The next -- as I work down this slide, the other thing we did, and Dr. Riccardella alluded to this earlier, is that we defined more explicitly what an LWR water environment was and changed throughout the NUREG reactor coolant to water with the notion that these effects aren't just seen in reactor coolant water but, really, any water, especially if it's high temperature water that could be throughout -- that could be located in a commercial LWR plant.

So we expanded the scope of the evaluation in that sense. And then the other thing we did, which we have done before but every time you do it you're always surprised at the volume of changes that occur is that we went through rigorous technical editing yet again. So and for those of you that have done this on a big document that causes you more consternation than you might expect initially. So be careful what you ask for. Okay. Any questions on the NUREG before I go into the Reg. Guide overview public comments?

Okay. So the Reg. Guide -- Revision 1 of the Reg. Guide went out in later 2014, November. The public comment period ended in early January of 2015. We got formal public comments from seven commenters. You'll see exactly the same form of the table that we used for the NUREG and we used the same form for tracking the comments except we didn't have subcomments from these guys. The comments were not quite as involved. That's the comments that we got on the NUREG.

We had 49 total comments. So still a large number but not nearly as large as we received on the NUREG. And it's worth noting that four of the organizations -- Rolls Royce, Westinghouse, AREVA and EPRI -- commented both on the technical basis document -- NUREG 6909 Rev. 1 -- and then also the draft guide.

And when you look at some of the comments they had -- they carried over comments they'd made on the technical document and made them again on the Reg. Guide. So they wanted to make sure that we heard them the first time.

Overview of those comments -- almost all of the comments were associated with the following areas, and I've tried to break it down somewhat statistically. Although as you might imagine, some comments are not easy to -- you know, it's not totally obvious where to group them but I've done the best job I can.

Almost half the comments with respect to the Reg. Guide were either editorial or clarification type of comments. There were 22 percent, a fairly large number, that commented again on the Rev. 1 technical basis. So a lot of these comments were similar to comments that we had received previously on the NUREG document.

We had comments related to the applicability of those earlier technical reports so there's earlier NUREG reports that I told you about. And also the applicability of earlier guidance such as the previous -- or the original version of Reg. Guide 1.207.

And then we got several comments related to the scope of the Reg. Guide, its use and applicability and then some miscellaneous comments.

So I said with respect to the NUREG comments the staff agreed with the office on all of those comments, not quite unanimous agreement with the Reg. Guide public comments.

We did fully agree with about half of the comments and partially agreed with the other -- with another quarter of the comments, and the most common areas of disagreement were in these two areas related to applicability of earlier technical reports and guidance and on Reg. Guide scope use and applicability, and I've got some examples of the comments and our areas of disagreement for the sample both of those areas coming up.

So here is a sample comment that talks about applicability of prior guidance and it says the draft guide does not clarify if the use of 6909 Rev. 0 formulas remain accessible.

Several license renewal applicants have used these methods and formulas for computing Fen and would not risk to revise them just in order to meet Revision 1 criteria.

So we agreed with -- we disagreed with them that we should put guidance within the Reg. Guide. We wanted the Reg. Guide to be a summary of what we thought the current staff position was on this particular technical topic.

But we did agree that there needed to be guidance within the appropriate documents. So we did try to clarify at least in the response that, look, this is a Reg. Guide so it's -- A, it's not a requirement -- it's guidance, of course, and that any prior method or evaluation that's been approved by the staff remains valid.

We are not going back and trying to walk back the calculation of anything that a licensee has done. But we did agree that within the SLR-specific guidance we needed to have a clarification statement on the use of prior methods for SLR.

We have -- so even though we disagreed with the comment we recognized that we needed to add something or have something in the SLR guidance documents, which we do now, and I touched on that earlier.

In fact, if I go back -- excuse me for going back, just to remind you -- this is what we added in this middle bullet bolded in the SLR guidance document and that we said yes, you can -- you can use 6909 Rev. 0 or Rev. 1 or, and this is always the out, NRC-approved alternatives. So this last one opens the door. If someone really had an old analysis that they thought was still appropriate and applicable and they came in to the staff and they justified it and we agreed with them, then that's perfectly acceptable.

So we are never telling someone that they have to use this particular method.

MEMBER RICCARDELLA: But wouldn't the old analysis only be up to 60 years and the new analysis has to take it from 60 to 80, doesn't it?

MR. TREGONING: Yes. Yes. So when they come in to SLR they have to demonstrate that they are good out to 80.

MEMBER RICCARDELLA: Yeah.

MR. TREGONING: Now, if they want to make a case that they can do it with the old guidance simply acting on whatever number of cycles they need them to get for the component out to 80 -

MEMBER RICCARDELLA: Yeah.

MR. TREGONING: -- they are certainly free to do that. We would rather they not do it. We'd rather they use the current guidance. But, you know, we are flexible. We will evaluate whatever is submitted to us and see if it's acceptable or not.

MEMBER RICCARDELLA: Uh-huh.

MR. TREGONING: So I apologize for cycling back so far. Let me get us back to where we were. Okay.

So this next comment talks about scope used and applicability of the Reg. Guide and there is two sort of important points here, and I'll paraphrase this comment.

This was a much longer comment from

Westinghouse so I've just tried to capture the -- sort of the most salient points.

And the first point is, you know, it states in the Reg. Guide that these methods apply to components exposed to reactor coolant that are required by regulation to have a fatigue CUF evaluation or have an existing CLB fatigue CUF evaluation.

And it goes on to say that there are components that have an existing CLB CUF evaluation in secondary systems. They are not required by regulation to have a fatigue CUF. The applicability of Fen in such components should be clearly stated.

So, you know, we agree that and we tried to clarify those applicability statements in response to this. So, you know, the points I'd like to make is that the Reg. Guide is applicable to both primary, pressure boundary or secondary systems.

We did agree that the draft Reg. Guide did not clearly define and use the terms reactor coolant and coolant with water. We went back and we replaced that and we also added a definition for what LWR water environment really means for the use of this Reg. Guide. And then we also went back and tried to clarify the Reg. Guide applicability and this is something that's changed even since I came and talked to you guys at the subcommittee meeting because we were having still additional dialogue with respect to the staff.

So if you look at the final guidance that's in front of you for review and approval, what it states with respect to applicability is it's for licensing actions associated with the following three actions: one, reactor design submitted for NRC approval or -- I am sorry, reactor design submitted for NRC approval.

That's the first application. The second application would be for operating reactors pursuing license renewal and the third application would be for plants where addressing such effects is part of their current licensing basis. So that third bin is meant to catch those other areas where it's part of their regulations that they have to address a specific component.

So getting back to the commented, if you're not required by regulation to have fatigue CUF you don't have to use those guides but you could. If you're looking at a fatigue evaluation for asset management purposes and if I was doing an asset management calculation I'd want to know what the environmental effects were to know if I needed to replace or repair a certain component after a specific time. You could use this guidance if you wanted. But you wouldn't be required to.

MEMBER SKILLMAN: Rob, before you change, just hold that thought. There are times when station staff is driven to take action not because of what is occurring here in the NRC but what is occurring, if you will, with ANI -- American Nuclear Insurers or the Hartford Steam Boiler.

In the case where you might or in the case where an analyst might be giving consideration to a change concerning the 5059, for instance, on a steam blind which would be ASME section three request to probably seismic one, depending on where it is. How is the steam environment treated because it's not air and it's also not water?

MR. TREGONING: Yeah. Steam environment wouldn't be applicable.

MEMBER SKILLMAN: Would not be applicable?

MR. TREGONING: Because it's not a water environment.

MEMBER SKILLMAN: Oh.

MR. TREGONING: Yeah.

MEMBER SKILLMAN: Thank you.

MEMBER KIRCHNER: Could you clarify? The very last bullet on this slide --

MR. TREGONING: Yes.

MEMBER KIRCHNER: -- it must be -- this must be a typo. I just don't think I understand it. Would you try it?

MR. TREGONING: The last bullet?

MEMBER KIRCHNER: Yeah.

MR. TREGONING: Yeah, it might be a little bit of a typo. I think I've worded it -- so plants where they have to consider where parts of their current licensing basis requires them to consider environmental effects in their fatigue evaluation. That's maybe a better way to say that.

MEMBER KIRCHNER: Requires their current license.

MR. TREGONING: Yes. Yeah. Either something in their tech specs -- you know, maybe they've come in to us at some point in time and said hey, I am doing this and it's part of my tech specs or I am required to consider fatigue or consider the environmental effects as part of my fatigue evaluation.

To be honest, I don't know how many plants might fall within this third category.

MR. STEVENS: Rob, wouldn't it be any plant that's been approved for 60 years?

MR. TREGONING: Well, potentially, again, but -- yes, potentially if they were approved for 60 years and as part of their license -- their first license renewal they did -- they considered water effects and that made it into the -- you know, the staff safety evaluation.

And yes, that becomes part of their current licensing basis at that point.

MEMBER RICCARDELLA: But you've already said it's okay to do that with Rev. 0, right?

MR. TREGONING: Yeah. So right, that's what I am -- right. So they don't have to redo that.

MEMBER RICCARDELLA: Yeah.

MR. TREGONING: But what would happen if they come -- well, let me give you this. If they came in for another application -- let's call it a power operate, right, so it's not a subsequent license renewal, right.

They would have already done a water effect evaluation as part of their license renewal evaluation. So then they would have to also look at that evaluation potentially on what the effects of the power upgrade would do to that evaluation.

So that's not specifically mentioned here but that's one thing I can think of off the top of my head that might sort of lead -- have a plant end up being in that bin. Do you agree with that, Gary?

MR. STEVENS: Yes.

MEMBER RICCARDELLA: Do you understand it? Because I don't.

MR. STEVENS: Well, I think -- I think the point would be -- what comes to my mind kind of -- maybe I'll try and say it differently but the same as what Rob just said, that if I went to 60 years and I -- and I had -- and I evaluated environmental effects and got approval for that, how far to my CLB? So if I now come in and do, you know, measurement uncertainty recapture, power upgrading, anything like that, it would make subject to addressing those effects.

MEMBER RICCARDELLA: And that would be required to be done by Rev. 1 instead of Rev. 0?

MR. TREGONING: Wouldn't be required. None of this is required.

MEMBER RICCARDELLA: That's right. Reg. Guide is not a requirement. Yeah. Okay.

MR. TREGONING: The guidance would be use Rev. 1, right. But, again, the licensee could come in and make whatever case they deem would be appropriate for that evaluation and we would review and evaluate.

MEMBER RICCARDELLA: Uh-huh.

MEMBER CHU: Can I -- can I ask a question? I don't know if it makes sense or not. Can you go to a reactor that's being decommissioning -- decommissioned to look at some of the metal components and validate some of the fatigue data?

MR. TREGONING: See, I think we need to hire you in research because we are always looking for good ideas like that. So yeah, you know, and I'll put in a plug for this.

We are actually having -- in early March along the same time that you guys are meeting next we are having what I am calling a harvesting workshop and we are looking at -- because of the number of plants that are decommissioning there is a lot of opportunities to get ex-plant materials and -- but those things are costly and require a lot of planning. So we are actually trying to develop a way to proactively assess, you know, how can we get the most bang for our buck -- what technical issue is most challenging that we can only address by getting -- either evaluating components that have been in service or getting those components and testing them further.

So that's actually the focus of this workshop that we are having on March 7th and 8th. So I'll put in a plug for it right there so thank you for that.

MEMBER RICCARDELLA: March 7th and 8th?

MR. TREGONING: Yeah. It's actually -- it's actually going to be at NRC and it's a workshop that's going to be NRC as well as EPRI and DOE. We are sort of collectively sitting down and it's meant to be a workshop so it's going to be very informal, definitely working level.

But the whole notion is to try to come up with a better -- we have done -- and I am way off topic so I apologize. I'll get us back on. We have done harvesting in the past very ad hoc -- oh, this plant's decommissioning -- let's go grab some materials.

We want to be more proactive in the future -- plan better, do it more efficiently, hopefully get high value information for less money than we have spent in the past. That's the whole notion for that slide. Let me bring it up.

MEMBER SKILLMAN: Well, I think this is a good -- a good topic for another reason. Is that effort aimed at a particular target such as validation or exploration of SLR information?

Because there are seven or eight topics that are open in SLR that are highly important to what we will be talking about in March and April.

MR. TREGONING: Yeah. I think -- you know, SLR is an obvious example because, you know, you talk about extending lives out to 80 years. But, you know, NRC is in the safety business.

So whether it's SLR related or current licensing related, it doesn't matter. So I think a lot of what we are looking at will end up having some applications to SLRs.

And again, we are looking at expanding this to -- not just metallic components but things like cables and concrete and other passive systems that we have aging management programs designed for.

And then we have a lot of good test information but it'd be nice to benchmark it as well with actual service evaluation of some actual components. So yes, we are trying to -- SLR will be a big customer of this particular activity but I like to think broader than that. Really, anything that we are dealing with I think is -

MEMBER RICCARDELLA: But I think as far as harvesting, I mean, we have a lot of what I'd call real problems out there that where, you know, degradation has occurred in plants that would be -- I would say would be better candidates for harvesting. This is more of a theoretical component.

I mean, we have never really had, to my knowledge, a failure in the industry because we use an air fatigue curve instead of a water fatigue curve, you know.

It's really -- there is laboratory data that says it's not -- you know, that says the air curves may be nonconservative in some cases. But I am not aware of any real field incident.

MR. TREGONING: Oh, we have had a lot of fatigue incidents in the field.

MEMBER RICCARDELLA: They have?

MR. TREGONING: Now, whether they were -- what components were due to the environment versus lack of understanding of actual conditions compared to design conditions, I mean, I think that's open to some debate.

MEMBER RICCARDELLA: Okay.

MR. TREGONING: You know, so, yeah, there is certainly been -- we certainly have good operating experience on fatigue incidents and in many cases we have gone in and we have provided guidance over the years and you've been -

MEMBER RICCARDELLA: I mean --

MR. TREGONING: -- you've been part and parcel of developing a lot of that guidance.

MEMBER RICCARDELLA: Most of the cases -- but most of the cases I am familiar with though are because we missed the loads, right?

MR. TREGONING: Yeah.

MEMBER RICCARDELLA: And we just didn't anticipate the loading conditions -- thermal stratification and things of that sort as opposed to the fatigue curves being inadequate.

MR. TREGONING: Well, but that's -- so that's the big -- that's the oh, I am not even in the right ballpark question, right, because I didn't even consider this whole class of loads.

So if you can't get that right then yeah, that's obviously going to be a big player. So but now I think -- as we have learned from that now we are getting into the next effects of okay, now you've got the loads right or at least the major load players. Now I think these other things can become important as well. So --

MEMBER RICCARDELLA: Well, there is this big thing that -- you know brought up and it's the difference between crack initiation and crack growth because what you're talking about here with these fatigue curves isn't failure because you're talking about a little tiny specimen.

And, you know, when you hit the life on that curve that corresponds to crack initiation, you know, and that crack initiation might be in an eight- or nine-inch thick nozzle and that doesn't mean that you -- you know, you're at end of life.

It means you just potentially have crack initiation.

MR. TREGONING: Yeah, and in the NUREG we tried to be clear what we think exceeding CUF means and we said, basically, the formation of an engineering-size crack, about a three millimeter crack.

After you have a three millimeter component that could be failure. If you have an eight-inch component, yeah, you got an 80-bit crack. So you have to look at your application to see how much remaining life you have, given that initial crack.

Okay. Let me move on. I am getting close to the end of my time. So I don't want to shortchange -- I've only got a few slides left. So this next slide provides and overview and I think I've touched on these are the changes.

So I mentioned already we change reactor coolant from coolant to water throughout the Reg. Guide including the title. We have clarified the applicability. Maybe it's still not clear enough. But I already went over that.

I also -- we also clarified that at least for the nickel chrome alloys that 718 is not applicable for use of the Reg. Guide just because we didn't have enough environmental data and, again, we just clarified that the nickel alloys should use the stainless steel design curves. They could either use Rev. 1 or the associated ASME curves.

So that brings me to my last slide in terms of status and next steps. So Revision 1 of the Reg. Guide we completed the technical concurrence about the same time as we had the subcommittee meeting, and I apologize for being a little bit outside of process on that. That was really driven by the fact, again, that we could get these documents out in concert with SLR guidance documents and the December subcommittee meeting was the only meeting that you guys could accommodate us. So, again, I apologize for being a little bit out of process with that -- with that regard.

So, again, we have had some changes since that subcommittee meeting and I think Chris has provided you with a red line strikeout. No technical changes at all, several editorial changes which I am sure you've seen, and then these applicability statements that are in Part B and C.

One of the things in the draft guidance we didn't have the same statements in Part B and C. Now if you look in B and C they are the same statements and they are meant to capture the applicability that I tried to summarize here in the -- in the -- in the presentation.

With respect to the technical basis document 6909 Rev. 1, there is still some technical editing changes that I am in the process of incorporating.

MEMBER RICCARDELLA: I thought that was already issued.

MR. TREGONING: No, it has not been issued yet.

MEMBER RICCARDELLA: Oh, okay, because the version I have says published December 2016.

MR. TREGONING: Yeah, that's -- okay. That was -- that was, I guess, our hope that it would be published. That was a hopeful statement. That probably should have been blank.

MEMBER RICCARDELLA: So but you're not looking for us to comment on that -- on the NUREG or just --

MR. TREGONING: We are not asking for you to comment. Of course -

MEMBER RICCARDELLA: No, no, I --

MR. TREGONING: -- because given your liberty you can comment on whatever you feel -

MEMBER RICCARDELLA: I was under the impression we were only comment -- or giving our opinion on the Reg. Guide, not the Reg. -- not the NUREG.

MR. TREGONING: That's all we are asking for but recognizing that the Reg. Guide basically says Appendix A in the NUREG. You know, that's really the guts of the Reg. Guide. So recognize that those things go hand in hand.

MEMBER RICCARDELLA: Not Rev. 1. Not Rev. 1 but Rev. 0 maybe back in 2007, right.

MR. TREGONING: And then again as -- there is a few -- you know, as it's gone through technical editing I need to make a few responses to the public comments because some of the public comments give quotes from the NUREG that says this is what we changed, quote, and when the technical editors got a hold of it some of those quotes changed. So that's going to be -- I need to make sure that the quotes are indeed accurate. So that's going to result in a few minor changes in those responses to public comments. Bu, again, nothing substantive.

So as I mentioned previously, we are requesting your recommendation to finalize the regulatory guidance and the plan has always been that we finalize it before issuance of the SLR guidance document and their -- if you look at their schedule they are planning for mid-2017. And we are hoping if all goes well with the process and there is no hiccups that we could finalize the Reg. Guide and the NUREG because, again, we want to finalize them jointly about the end of next month.

But given that I said December, you know, that could still be a moving target. So that was all I had. Is there any other questions?

MEMBER BLEY: None for you, but Pete, in our subcommittee and here as well we talked to you an awful lot of what's in the NUREG and well, we can talk among ourselves about whether the letter should cover both in one. But I mean, it's essentially the same.

MEMBER RICCARDELLA: Okay. Yeah. Okay. Thank you very much for all those. A very good summary presentation.

At this time, could we get the -

MEMBER BLEY: Just ask for comments and they should have --

MEMBER RICCARDELLA: We'd like to ask for any comments from the public. Is there anybody out there who would like to comment? I see -- I see nobody in the room so I assume we have no public comments on this.

MEMBER BLEY: We don't have the line open. Is it open? Is the bridge open? Okay, thanks.

MEMBER RICCARDELLA: Okay. With that, I'll turn the meeting over to -- back over to our chairman.

MEMBER BLEY: Thank you very much, and substantially early. We will reconvene at 2:45 to take up the generic quality assurance lessons learned issue.

We are recessing until 2:45.

(Whereupon, the above-entitled matter recessed at 2:23 p.m. and resumed at 2:47 p.m.)

MEMBER BLEY: We're back in session and I will turn the meeting over to Harold Ray to continue.

MEMBER RAY: Thank you, Mr. Chairman. A few remarks I'll make here to put things in context. I won't promise not to say other things during the presentation but thought I'd get some of them on the record here.

Now, almost a year ago the committee reviewed five exemptions to the AP 1000 design certification which were required: to enable the certified design to perform intended functions.

Now, when an exemption to a certified design is needed in order for it to perform intended function, there is also a need to identify any other examples and define what steps must be taken to minimize such occurrences in the future.

This is referred to as an extent of condition and lessons learned exercise. Everyone involved in AP 1000 -- the NRC, the design certification holder and the COL holders -- engaged in this exercise at that time and we concluded in our letter that the effort had been satisfactory.

So today we aren't here to revisit specifics of the AP 1000 experience. I don't need to point out that the inability to preform intended functions under off normal conditions often isn't self-revealing under normal conditions and for that reason programmatic quality insurance requirements applicable to all safety-related activities have been adopted in the regulations and elsewhere as in the ASME code.

Now, this is an informational meeting which responds to our recommendation in the AP 1000 letter -- that the generic lessons learned relative to the design process leading to certification, and I want to underscore that phrase there. The design process leading to certification should be identified and further evaluated. In expressing the recommendation this way we recognize the limitation which necessarily exists relative to who is and who is not yet an applicant and therefore subject to regulatory compliance and oversight.

In informal meetings with staff to prepare for today's presentation a number of questions were posed for their consideration, and these are identified on slide 15, and the staff has provided references there to where the responses are provided in their presentation that we will hear now.

And finally, as I noted, this is an informational meeting and although it's unrelated to the purpose of this meeting I want you to be aware that at next month's P&P we will discuss what interest the committee may have in review of proposed Revision 5 to Reg. Guide 1.28, quality assurance program criteria for design and construction.

So we will have a similar sounding discussion perhaps and this is for the purpose that I stated, and with that I'll turn it over to Kerri or Jeremy, whoever's going to take a lead here.

MS. KAVANAGH: Well, thank you for inviting us to come and present. My name is Kerri Kavanagh. I am the chief of the quality assurance vendor inspection branch.

It's my pleasure to introduce you to Mr. Jermaine Heath, who has been assigned to present our answers to today's ACRS questions. Be nice to him. This is Jermaine's first time in front of the ACRS. He --

MR. HEATH: Oh, no. That just gives --

MS. KAVANAGH: Yeah. He is -- he is well prepared and he has a very good presentation for you and I hope you come away from this briefing, you know, better informed and prepared to look at several of our documents that are coming before the ACRS in the next several months.

And with that, Mr. Heath.

MR. HEATH: All right. Good afternoon.

Again, my name is Jermaine Heath. I am a reactor operation engineer in the quality assurance vendor inspection Branch Three of the Division of Construction Inspection and Operational Programs of NRO. I am pleased to address the ACRS staff this afternoon.

We are here today to present on QA program implementation under 10 CFR Part 52. But first I want to begin by giving a brief introduction to the responsibilities of the quality assurance vendor inspection staff.

There are three quality assurance vendor inspection branches located within the NRO and they make up the vendor -- the vendor inspection center of expertise.

Quality assurance vendor inspection branches lead and perform routine and reactive inspections and we also conduct QA implementation program inspections for new and operating reactors.

Our branches also perform QA program implementation reviews for Part 50 and Part 52 applicants and also for initial test program applicants and we also provide support for NRC's Region 2 for new construction activities.

We have -- the QUIV staff has been tasked by the EDO, as Mr. Ray said earlier, to give our presentation to the ACRS staff on Appendix B application to Part 52. We received a list of questions from ACRS. We will be addressing those questions throughout the course of this presentation. As Mr. Ray said, those questions are attached in your supplemental slides and I will be making reference to those questions as they -- as appropriate -- as I go through the slides. So feel free to ask questions as we go along.

As walking through the presentation, I'll give a little bit of background to why we are here. Then I'll jump right into the QA program implementation for new reactors.

I'll be talking about the DC and COL responsibilities to Appendix B. Then I'll cover the NRC's QA oversight which includes the QA licensing portion plus our inspection programs.

I'll give some concluding remarks and then there will be a time for any additional questions that you may have.

So I'll be brief here. Mr. Ray gave a pretty good introduction synopsis of kind of the background here. But last April ACRS reviewed several exemptions for which Duke Energy included in its Levy

County COLA for the AP 1000.

We not intend at this briefing to go into details about all the exemptions and their associated departures.

The ACRS staff recommended that we evaluate on a generic basis one of their any lessons learned relative to the quality assurance program implementation during development of design seeking certification under 10 CFR Part 52.

So the NRC quality assurance staff conducted an evaluation based on the ACRS recommendations and we are here today to present our findings and address your questions related to these items.

Is everyone keeping up? Usually I want to say next slide and I know I am going through them but do I need to call out where I --

MEMBER BLEY: Just march ahead.

MR. HEATH: March ahead. Okay. All right. So as you may be aware, back in 1984 the NRC conducted a study which became known as NUREG 1055.

This study was a lessons learned review of the underlying causes of major quality assurance issues identified during design and construction of power plants in the Part 50 process.

The study was important because NUREG 1055 was a catalyst to what we -- to what are now the current requirements for new reactors under 10 CFR Part 52.

The NUREG study determined in a root cause that the major quality related problems associated with design and construction back when was the failure or inability of some utilities to effectively implement a management system that ensured adequate control over all aspects of the construction project. That include ineffective QA implementation and also poor contractor oversight.

Additionally, the study found that the NRC's past licensing and inspection practices did not adequately screen construction program applicants for overall capability to provide effective management oversight over the construction projects.

So QA lessons learned from NUREG 1055 were incorporated into the current Part 52 licensing process and resulted in several changes to the NRC's inspection manual chapters and our inspection programs.

Since then the NRC has become more engaged by conducting more QA inspections along the design certification review process.

MEMBER RAY: Let's just stop there for a second, Jermaine, and think about, again, we are not here to discuss what happened in AP 1000.

But when you say current processes, do we believe that the experiences that we had with AP 1000 predate the more QA inspections during the DC process that you're referring to?

Or were they subject to the more QA inspections during the DC process that you're referring to there?

MR. HEATH: So I think what I'd like to say I think the rigor that are in the current Part 52 licensing processes now have been beefed up before based on the NUREG from what we were doing in the NUREG study. The lack of -- it was a lack of oversight and inspection -

MEMBER RAY: Correct.

MR. HEATH: -- and then attention to detail during the license and review process.

MEMBER RAY: I understand that point but I am now asking not before 1055 and after but I am asking were these enhanced inspections for QA program implementation the same -- are they the same today as they were during the time when the design work on AP 1000 -- which is a long time ago, but not as far back as 1984 -- were taking place. That's --

MS. KAVANAGH: So the answer is that the procedures that we use now for review and approval and inspection have stood up when NRO was stood up in 2007.

Before that, we were using procedures that were put in place as a -- as a learning curve to the new office. So the answer to your question is it's more robust now than it was before NRO was stood up.

MEMBER RAY: Okay. That's the question. Okay. So we don't know this to be true but one could assume or believe, since we go all the way back to AP 600 with some aspects of the issues that were in hand that they predate when NRO -

MS. KAVANAGH: Right. Right.

MEMBER RAY: -- the time when you're talking about.

MS. KAVANAGH: For AP 600 the NRC was performing inspections of design and testing for AP 600. It was part of the process. It's just -- I don't think it was as formalized as it is now.

MEMBER RAY: Yeah, and, of course, we are -- and I'll get to you in a second -- we are talking about QA, not just inspection of work in the shop or whatever.

MS. KAVANAGH: Right.

MEMBER RAY: Okay.

MEMBER RICCARDELLA: What was the date of this NUREG again?

MEMBER RAY: '84.

MEMBER RICCARDELLA: 1984.

MR. MCINTYRE: Yeah, Mr. Ray, back in -- we started design qualification test program inspections and design reviews back to the AP 600 days and we were doing a lot of those inspections up in the old Monroeville and they weren't fabrication type activities.

Those were implementation of the design, control, corrective actions -- the whole Westinghouse quality assurance program.

So that predates the office of new reactor work that Kerri was talking about.

MEMBER RAY: Well, I understand, and based on what you just said I would say well, then perhaps we aren't any stronger now than when these events occurred in the AP 1000 experience. But that's a presumption or assumption that one would make, perhaps. I don't know. Kerri, did you want to say something more?

MS. KAVANAGH: All I was saying is that,

you know, the NRC does a sampling and we do -- we do what the -- the most significant systems that we believe that need to be looked at.

MEMBER RAY: Let me ask one other question, since we are talking about Part 50 as well as Part 52 in this historical context. Would it be accurate to say that the agency expects COL or the ultimate operating license holder to do more than what the agency itself does -

MS. KAVANAGH: Absolutely.

MEMBER RAY: -- at the point in time when there isn't a COL relationship?

MS. KAVANAGH: Yes.

MEMBER RAY: Okay. Thank you. Go ahead.

MR. HEATH: So now I want to talk about the Appendix B responsibilities of the DC and COL applicant under the Part 52 process.

ACRS posed a question to the staff: Does 10 CFR Appendix B apply to the development of safety-related information reflected in its certified design.

So Appendix B includes requirements to establish and implement a QA program for any nuclear facility application. These QA program requirements apply to the siting, design, construction, operation and decommissioning of the nuclear facilities and all activities that could affect the safety-related -- quality of safety-related applications at these sites.

So yes, Appendix B applies to the development of safety-related information reflected in the certified design. Every applicant seeking a design certification or a COL referencing a design certification under Part 52 is required by the provisions of 10 CFR .47 and 52.79 respectively to include in its final safety analysis report a description of the QA program that was applied to the design of the safety-related structure systems and components.

Control of initial test programs, which the NRC evaluates under standard review plan 14.2 is also required to be included in an application.

Now, for DC applicants, the quality assurance program description, otherwise -- or otherwise a QAPD, must include a description of how the applicable requirements of Appendix B were satisfied.

A QAPD submitted by a DC applicant may be a topical report or a part of the final safety analysis or a part of the safety analysis report. A QAPD submitted by a DC applicant would include or would address design QA activities in support of the design certification but it would not address design and construction activities once construction begins.

But to that point, though, I would like to add that the NRC QA staff is involved very early in the DC review process and in most cases the QAPD is reviewed by the staff and approved before the DC application is even submitted.

Now, for a COL applicant, the QAPD also must include a description of how the applicable requirements of Appendix B were satisfied and how the QA program will be implemented to a level of detail consistent with the current industry standard which is NQA 1, which the NRC has endorsed or approved. But a QAPD submitted by a COL applicant would apply to all phases of the facility's life including design, construction and operation.

MEMBER SKILLMAN: Jermaine, before you proceed, I would like to explore that third rule a little bit more. It seems to me that those two activities -- the DC applicant and the DC applicant's activities and the COL applicant and the COL applicant's activities -- are fundamentally different. They are similar but they are different. Give you an example. I have a super whamodyne NSSS I am trying to get licensed in this country and I am the DC applicant. I come and I say, here are 21 or 22 chapters of how my reactor design will fit into Part 52 and be legal in this country, meeting all of the required codes and so on.

When the NRC publishes its SER and that SER is approved through the processes for which a DC applicant is intended to proceed through, that design is locked in because of finality rules.

Now, I am, as a DC applicant, really screwed up in Chapter 4. I made some major errors that neither you nor I detected. In my view, that does not become a COL applicant problem. That problem remains with the DC holder. But here's my question. When something like that is discovered, why doesn't that become a Part 21 issue requiring the reactor construction inspection branch to pursue a Part 21 instead of involving the COL applicant?

Because it seems -- and I go back to having worked for an NSSS vendor for a long time -- we owned the problems that were part of our NSSS design. I mean, we -- and we had to repair those on our tab. And so my real question is why aren't issues that bleed out of the DC application Part 21 issues? I am a supplier.

MS. KAVANAGH: Right. So are we talking

about you identified this issue after it's been finalized -- after we give you a --

MEMBER SKILLMAN: Yeah. Yeah.

MS. KAVANAGH: So, I mean, that's a good question because OGC has issued a generic letter or a risk on this particular issue but I believe it was for design certifications for -- that have not been certified.

And so as you're going through the process as to why Part 21 would apply to that particular NSSS as they are developing the design and making sure that the staff is notified for the different versions. But at -- and I am not sure we have the answer for you but at the time that finality is reached you no longer own the design.

MEMBER SKILLMAN: Bingo. That's exactly what I am saying.

MS. KAVANAGH: So you wouldn't -- Part 21 wouldn't necessarily apply to you.

MEMBER SKILLMAN: Oh, I know. Once I have -- once I have a certified design and it becomes an appendix to Part 52 --

MS. KAVANAGH: Yep.

MEMBER SKILLMAN: -- okay, actually it's owned by the public.

MS. KAVANAGH: Yep.

MEMBER SKILLMAN: Okay. And let's say we find a major issue in that design cert. Who owns that? And I am asserting it's owned by the DC applicant because the DC applicant that produced that design certification is the one, at least in theory, has the analysis, the databases, the configuration control and should have had 18 parts of Appendix B striped right on the front of it because that's what they submit under.

MR. HEATH: So if the issue is identified by a COL applicant I assume that's who my -- is identifying this issue?

MEMBER SKILLMAN: Could be. Could be identified by ACRS. Could be identified by a new staff member who shows up and says, hey, wait a minute -- there is something peculiar here.

MEMBER RAY: Could be identified by a English safety review.

MEMBER SKILLMAN: Bingo. You can see where I am going and here's why I am doing this. I am trying to -- I am trying to get it clear out of my mind between the DC applicant's responsibilities and the COL applicant's responsibilities. I think I understand them.

MS. KAVANAGH: Right.

MEMBER SKILLMAN: But to me the -

MEMBER RAY: Can I interrupt you for just a second? Just trying to simplify this. We all, I think, are familiar with the fact if I am a pump vendor and I have a defective -- a deficiency in the pump that I am providing to the industry I have an obligation under Part 21. It's not a certified pump but it is a complement that's being supplied to the industry.

Tell us the difference between that circumstance that we all are familiar with and the circumstance that Dick is describing in which, well, the pump has now become a certified design for a reactor but otherwise it's still the same.

I mean, I am the vendor and I am selling it to people and I discovered this mistake in it. I think he's asking why doesn't that trigger a Part 21 just like it would be for a pump vendor.

MS. KAVANAGH: That's a good question and I don't think I have the answer for you. I know there is a -- there is a methodology defined in Part 52 as to how you would change the design to correct something that you've identified. At a minimum, we would expect that you would put that into your corrective action program and your design control program to be able to submit that information to the staff to make that change under Part 52 but neither Jermaine or I or the Part 52 expert on how you would change the design. So we could take that question back for you and try to get you an answer.

MEMBER RAY: It's more a matter of reporting it as opposed to the execution of the change that I think we are talking about. But let's do note what you said, which is we think we understand that question. We don't have an answer right away and for the sake of time let's proceed with that as we will decide what to do about it at the end.

MS. KAVANAGH: Okay. Fair enough.

MEMBER SKILLMAN: Thank you.

MS. KAVANAGH: Uh-huh.

MEMBER SKILLMAN: Thank you.

MR. HEATH: Okay. So on this slide simply highlighting some very relative and important sections of Appendix B, requirements as they apply to both the DC and the COL applicant.

ACRS posed the question to the staff, who is responsible for verification that Appendix B requirements have been met for information reflected in the certified design and how is compliance with

Appendix B verified prior to NRC approval of the certified design. And I'll revisit the latter half in a -- in a later slide. But we have just established that Appendix B applies to the development of safety-related information reflected in certified design. This is the reference to 52.4 -- 10 CFR 52.47. But I want to highlight criterion one of organization, Appendix B.

So who is responsible for the verification that Appendix B requirements have been met for information in the certified design? Well, very clear in criterion one, the applicant is responsible for the establishment and execution of the quality assurance program.

Now, the applicant may delegate parts of the work that establish and implement the QA program to other entities such as the contractors or suppliers. But the applicant shall remain overall -- shall maintain overall responsibility for the program.

Now, if the applicant chooses to contract portions of the setup or the implementation of the QA program out, the applicant must describe the extent to which this work is delegated.

Under criterion three, also required is the organization must establish rules and responsibilities for prepare and reviewing and verifying design documents.

These responsibilities apply to all aspects of the design including specifying design inputs and output data, their analysis establishing acceptance criteria. It applies to design drawings and then any implementing procedures.

Very important, I'd like to point out criterion 18, which is audits. Very important. Criterion 18 requires the DC or COL to establish an audit program.

This includes both internal and external audits. For internal audits the applicant must conduct audits of its internal controls to verify implementation of the QA program complies with the requirements of Appendix B. For external audits, and this encompasses criterion 7, which is control of purchased material, equipment and services. The organization must also establish provisions for auditing its suppliers and contractors.

Where these audits are used they should be routine at a frequency that's based on the complexity of the work. Typically, we see three years. That's about a standard.

So the COL applicant is responsible for

verification that Appendix B requirements are being met by its suppliers of safety-related parts and services. This would include the design authority, who would be the DC applicant in this case. ACRS --

MEMBER RAY: Excuse me. That was a little bit of a confusing comment. But -- do I understand what your intent is that -- because the question went to oversight, not execution. Oversight.

That the COL applicant is responsible for oversight of its vendor, which is -- includes the DC holder once the COL applicant has a relationship with that vendor.

MR. HEATH: Yes.

MEMBER RAY: But prior to that time, which is the time period during with the certification takes place, the only oversight that can be provided is that which occurs after the application has been docketed by the folks you guys represent.

MS. KAVANAGH: Yes.

MR. HEATH: That's true.

MEMBER RAY: All right. So one of the things, and I want to move on and not get into -- bogged down because this can happen easily, but one of the issues then becomes as our COL holders for AP 1000 found how far back do I look once I -- if I take a certified design can I just say it's a certified design -- I don't have any further obligation, which is the logical position to take? Or do they have some other responsibility?

And I think it's the former. You simply say from here on I have oversight responsibility but I am not obliged to do oversight of what came before. Fair?

MR. HEATH: So that's a fair statement and it kind of leads me into the next question I am going to address --

MEMBER RAY: Go ahead.

MR. HEATH: -- which is to the translation of the design information.

MEMBER RAY: Yeah. Yeah. Go ahead.

MR. HEATH: So ACRS posed a question to the staff for safety related information developed by the design certification applicant.

One of the COL holder's responsibilities for verifying that Appendix B requirements are met for this information when implementing the certified design, again, I'll -- so I'll point to criteria 3 design control and I'll quote: measures shall be established to assure that applicable regulatory requirements and design basis information is translated into specs, drawings and instructions including provisions that the appropriate quality standards are specified.

So the COL holder is responsible for verification that Appendix B requirements are being met when translating information from the certified design into procedures while the NRC is providing an oversight function of suppliers of safety-related equipment and services through direct inspections in accordance with our construction inspection program, which I'll talk about here in a minute.

MEMBER RAY: We could spend a lot of time here but let's move on.

MR. HEATH: So, in summary, of course, all the activities I just described will be subject to the requirements of the applicant's corrective action program in accordance with criteria in 16 to assure that any issues resulting from the implementation of the Appendix B program are promptly identified and corrected. So -

MEMBER RAY: I like this figure, by the way.

MR. HEATH: You like this?

MEMBER RAY: Yeah. Yeah. I haven't seen it before so --

MS. KAVANAGH: It took him a long time to develop.

MEMBER RAY: Because we have to think about it again.

MR. HEATH: All right. Well, I hope you like what I say. So I talked about -- so now we talked about the DC and the COL applicant's responsibilities. Now I will talk about the NRC's QA oversight and response.

So we have established the applicant retains responsibility for the QA program. This is criterion one. The NRC is responsible for monitoring through oversight that the applicant is adequately implementing its Appendix B program.

Now, the NRC's QA oversight is an integrated approach achieved through the execution of two key components. There is the QA licensing reviews over on the left and then over on the right it's the construction inspection program.

The QA licensing reviews are conducted in accordance with standard review plan 17.5 in NUREG 0800 and the staff also implements the construction inspection program which addresses QA implementation through a series of inspection manual chapters and inspection procedures.

So ACRS posed a question to the staff, does a COL applicant have any responsibility for verification that Appendix B requirements are met for safety-related information reflected in the certified design.

And Harold, this kind of goes back a little bit to what you were talking about. So over here on the left QA licensing reviews, for a submitted DC application the NRC is responsible for reviewing that Appendix B requirements are being met for the safety-related information in the certified design.

MEMBER RAY: Now, that Reg. Guide 1.28, I'll point out to members, is -- the update of it what we will be talking about in the next full committee meeting in March.

MR. HEATH: So for DC and COL applicants there are typically 22 areas of a quality assurance program description that the staff will review. This includes all 18 criteria of Appendix B for which -- for which the applicant must specify QA controls.

The level of detail of how the DC and COL applicant will implement Appendix B is a combination of two things -- NQA 1 and Reg. Guide 1.28 -- where NQA 1 is the current QA standard that the NRC has approved that the DC and COL applicants use to establish an Appendix B program and it follows the 18 structure criteria of Appendix B. NRC Reg. Guide 1.28 endorses the NQA 1 standard.

Over here the bubble on the right side is the construction inspection with the other half that make up the program oversight. So in conjunction with the NRC's licensing reviews the NRC implements a rigorous construction inspection program containing various QA elements that verify that the DC and COL applicant is meeting commitments as set forth in their application.

The inspection program consists of a number of direct inspections of the applicant and also the applicant's contractors. The program is owned by NRO and implemented by NRO and the NRC regions. At this time, it's being Region 2.

MEMBER RAY: Okay. Now, just again to pause, we pointed out earlier but I -- let me emphasize here again, make sure that we are clear, this isn't just limited to inspections, which is one of the 18 criteria, but it's review -- I don't care what name you give it -- inspect, audit or whatever -- of the program.

MS. KAVANAGH: Uh-huh.

MEMBER RAY: It's not just the product.

MS. KAVANAGH: Right.

MR. HEATH: That's correct.

MEMBER RAY: Okay.

MEMBER SUNSERI: I've got a question. I am likely very confused on this point. So let me -- let me just go back a step here.

I always looked at Appendix B as applicable to licensees or applicants, right. So, you know, that's a class of nuclear industry stakeholder, right.

Suppliers that weren't a licensee or applicant but wanted -- that needed to implement a quality assurance program used NQA 1.

MS. KAVANAGH: Right.

MEMBER SUNSERI: So I am confused then to see NQA 1 up here in association with an applicant and what -- and what that tie or how that connection is made.

MS. KAVANAGH: All right. Do you want to address it or do you want me to?

MR. HEATH: Go ahead.

MS. KAVANAGH: All right.

MEMBER SUNSERI: So was my interpretation accurate of the division of authority between the two?

MS. KAVANAGH: Right. So Appendix B is

directly imposed on the applicant or a licensee. That's what it's applicable to. The staff, open since 1994, has found that NQA 1 is one suitable method to meet Appendix B and we have endorsed it in Reg. Guide 1.28, which you'll be discussing in a future meeting.

Suppliers have Appendix B or NQA 1. Generally, Appendix B is imposed on them via contract and one way for them to meet those requirements is by implementing an NQA 1 program.

So it's not -- Appendix B does not directly apply to them but it's contractually imposed.

MEMBER SUNSERI: So let me -- let me tell you how I've seen that practically implemented. So I am a supplier and I want to do work at a nuclear power plant.

MS. KAVANAGH: Yep.

MEMBER SUNSERI: If I don't have an NQA 1 program for my own operation then I am obligated to follow the client's Appendix B program and use their procedures and we are okay.

MS. KAVANAGH: That's correct. Yeah.

MEMBER SUNSERI: All right. But there is no such thing as a non-licensee or applicant that has an Appendix B program standalone, right? It's going to be an NQA 1 or it's going to be working for --

MR. HEATH: Don't assume.

MS. KAVANAGH: Oh, no. There -- yeah, there is suppliers still out there.

MR. HEATH: There is some suppliers that have Appendix B programs.

MS. KAVANAGH: NQA 1 is not the only method. It's just one method.

MEMBER SUNSERI: Okay. All right.

MR. HEATH: All right. So back to the diagram. You'll notice that the component on the left that is the QA licensing review is -- I've depicted it slightly smaller than the construction inspection program component. I did that for a reason.

This is because when a design is certified following the NRC's licensing review, the design is less than 50 percent complete, and this is in terms of design finality versus design implementation.

Design certification finality is part of the scope within that 100 percent of design completion whereas design completion entails the construction aspects. So this is after we certify the design.

So the QA licensing review is only part of the oversight process. The NRC provides oversight for the remainder of the implementation of the design through its inspection programs and it's this -- it's this implementation by our construction inspection program that provides a reasonable assurance that that the plant will be constructed and operated as required.

MEMBER RAY: Well, I'd like to minimize the confusion between the inspection of -- that's done through the inspection program and the QA program oversight.

I understand you do both. And by the way, let me clarify -- NQA 1 is the ASME QA program that actually was implemented after Appendix B to mirror Appendix B.

So if you've got a --

MS. KAVANAGH: Absolutely. It provides -

MEMBER RAY: -- if you've got a code stamp you've got to do NQA 1. But if you're doing, as we had a presentation on electronics beginning of this week, there is no reason why you would choose to use the ASME QA program. You might just as well use Appendix B.

MS. KAVANAGH: Right.

MEMBER RAY: Okay. But we -- I do find it problematic when we drift off into talking about how inspection verifies the implementation of a QA program because that's a very tough argument to make unless what you're inspecting is the QA program. If you're inspecting the product you're just inspecting the product and that's all you can say about it.

MS. KAVANAGH: Well, we based most of our inspection off of the QA manual. So --

MEMBER RAY: That's fine.

MS. KAVANAGH: -- and the implementing procedures, and then we will dive down into the design or whatever aspect we are looking at and bounce that back off their processes.

MEMBER RAY: Well, we have all had experience with vendor inspections and a lot of the inspection is also of the product itself.

MS. KAVANAGH: Okay. Fair enough.

MEMBER RAY: All right. So let's move on.

MR. HEATH: Okay. Same slide. ACRS -- it's another ACRS question. The ACRS posed a question to the NRC staff -- is there a transition of responsibility between the NRC review during the DC phase and the period when the COL applicant becomes responsible for implementing the certified design.

So there is no transition of responsibility between the NRC and the COL applicant. The NRC conducts the QA licensing review and the QA program implementation inspections for the DC applicant in support of that licensing review of certified design.

And I've said before the COL applicant is responsible for verification that the Appendix B requirements are being met when translating information from the certified design into specs drawings and procedures.

Furthermore, anytime the COL applicant initiates the purchase order for a safety-related -- for safety-related equipment services or services referencing the certified design, its COL applicant's Appendix B responsibilities will apply.

Legal authority -- ACRS posed the question to the staff, how is compliance with Appendix B verified for the period of time prior to a DC applicant submittal of their Part 52 application.

Again, the applicant retains responsibility for the implementation of the QA program. The DC applicant must apply Appendix B controls to any safety-related activity with the appropriate design reviews and implementing an independent QA audit program. This is a requirement of criterion 18 audits.

So for any information submitted in the

application under 52.47 the applicant must show how Appendix B was applied to the design.

Regarding the NRC's legal authority to inspect, the Part 52 formal application process begins with the DC COL application is docketed -- that is, when the NRC accepts the application for review.

For any time period prior to this docketing this would be considered pre-application phase. Currently, there is no regulatory basis to conduct pre-application inspections of a DC applicant's QA program.

However, for DC applicants most are well aware of the importance of establishing and implementing an Appendix B program prior to beginning to any work. In this case, they would submit their QA topical report early in the pre-application development stages.

COL applicants, on the other hand, cannot submit their QA topical early. Their QAPD must be submitted with their application.

MEMBER RAY: Okay. So the important thing to recognize here then is that this validation of the use of Appendix B during what occurred prior to docketing is an important part of the review that occurs after docketing and we just have to leave it at that.

One of the issues will be, and I am watching the clock here to make sure we get done in time, will be how aware potential certification applicants are of what rigor is involved in doing that.

You guys, did you say you had some public meetings or something in which people were being informed that they were going to have to pass this hurdle as part of the docketing process?

MS. KAVANAGH: Well, we have -- we have been asked to present Appendix B requirements to the advanced reactor, meaning that the next public meeting in April we will be making a presentation there.

We also hold biyearly workshops with our vendors about Appendix B Part 21 criteria applications. So every two years we are out there having workshops with the vendors, stressing not necessarily with the applicants but with our vendors on what the requirements are.

MEMBER RAY: Well, I hope they get the message because it's not something that's easy for them to have to do unless they are convinced they are going to have to do it.

MEMBER REMPE: So in those advanced

reactor discussions you hear a lot of these folks that are going to propose that they are ready for design certification wanting something earlier than a certified design. Have you been interacting with the office of new reactors or whatever they are called now and discussing with them if there will be any sort of QA requirements imposed on this earlier review that is being proposed over there? Do you understand what I am talking about?

MS. KAVANAGH: I am not familiar with this early review.

MEMBER REMPE: Well, they are talking about that they -

MEMBER BROWN: Who's they?

MEMBER REMPE: The advanced reactor designers would like --

MEMBER BROWN: DOE?

MEMBER REMPE: Well, they may be getting the money from DOE but no, they are separate companies that have reactor design and they get also money from perhaps venture capitalists or other folks with lots of money and they are saying oh, it's so hard to get a certified design -- we'd like something earlier to give the venture capitalists or whoever's sponsoring this in confidence. And so one question that might be interesting to discuss with the office of new reactors is will you impose any sort of QA requirements on them before the thing is docketed.

MS. KAVANAGH: Oh, well, yeah, they have the opportunity and the right to submit their QA program before it's docketed.

MEMBER REMPE: They have the opportunity. But if you're just wanting something that's a little cheaper to obtain they won't do it and that's what I am kind of asking, will the regulator try and encourage them to give us an idea of their QA earlier on.

MS. KAVANAGH: I can't speak to it. Joe, you want to introduce yourself?

MR. WILLIAMS: I am Joe Williams. I work in the advanced reactor and policy branch in NRO. I can tell you that we are encouraging all the prospective applicants to submit QA programs to us in advance and several of them have indicated that that is indeed their plan.

So if I understand where you were going with your question correctly, you were referring to what is sometimes called an incremental or a stepwise licensing approach.

And it would certainly be my personal

expectation. I think it would be the expectation of the organization that appropriate QA would be applied throughout the process including any incremental licensing product that was proposed and provided.

MEMBER REMPE: Thank you.

MR. WILLIAMS: You know, to give an example, we have discussed, for example, how provisions of Part 52 for standard design approval might apply and how that would go forward, and certainly a component of a standard design approval would include an evaluation of the QA program.

MEMBER REMPE: Thank you.

MEMBER KIRCHNER: And that would be NQA 1 -- based on NQA 1? That's the expectation?

MS. KAVANAGH: It doesn't --

MR. WILLIAMS: I, frankly, don't know.

MS. KAVANAGH: It doesn't have to be.

MEMBER RAY: No. NQA 1 really only has to apply to a code stamp holder.

MEMBER BLEY: For those interested in the issue that Joy just brought up, be sure to come to the subcommittee meeting the first week of March where we are going to talk about these things in some detail.

MR. MCINTYRE: Kerri?

MS. KAVANAGH: Yes.

MR. MCINTYRE: Let me try to simplify something with the -- I sense a confusion between Appendix B and NQA 1. Let's always remember that Appendix B is the umbrella QA program. Licensees chose to submit NQA 1 programs because it gave them the opportunity to take the part one and the part two requirements. Licensee QA Chapter 17 always said Appendix B and how to implement through all the ANSI daughter standards.

So NQA 1 took the -- all those daughter standards, made it a part two document and put it into the -- into one document together so it was easier for licensees to implement that one program as opposed to having Appendix B and 17 different supplemental commitments. So that's all in NQA 1 and we now endorse as part of Reg. Guide 1.28 part one and part two. So that's the beauty.

But everything still is Appendix B and there is -- you can implement an Appendix B program without implementing NQA 1 except what Mr. Ray said. Unless you're an ASME certificate holder then you're required to have an NCA 4000 NQA 1 program.

MEMBER SKILLMAN: Let me ask this. Your second bullet -- is that an invitation for a change for how the process is conducted? It would seem that anybody who would come in and say I really want to get a design cert would already be of the mind set to see an obligation to meet part B or Appendix B in 10 CFR 50.

So at least to me it's almost intuitive. If you're going to come in and ask for a design cert approval you will have already have needed to have instituted a program that you can defend.

MS. KAVANAGH: Right.

MR. HEATH: That is correct.

MEMBER SKILLMAN: And so --

MR. HEATH: But we are just talking about, again, our -- what stands as our current legal authority.

MEMBER SKILLMAN: I understand. But what I am suggesting is, is that something that should be marked as right for change, going forward? Because as Joy points out, we may have two, three, five different new designs coming in where is this is resolved on the front end there is clarity in how those applicants will move in this regard.

MS. KAVANAGH: Right, and I'd like to say something very tongue in cheek. But I don't think rule making is going to happen on Appendix B anytime soon, to be honest with you.

MEMBER SKILLMAN: That would be --
MS. KAVANAGH: It would have to be.
MEMBER SKILLMAN: -- it would be -
MS. KAVANAGH: Yeah.
MEMBER SKILLMAN: -- it would be handled on Appendix B.
MS. KAVANAGH: Yeah, and so -
MEMBER SKILLMAN: Okay. Fair enough. Just asking the question.
MS. KAVANAGH: Yeah. I mean, the regulation is its design and that design means when you're actually doing the design prepping for that application all that design work. However, Appendix B applies when you're an applicant. So -
MEMBER RAY: Well, Appendix B greatly predates Part 52 in principle.
MS. KAVANAGH: Agreed, and remember Part 52 always points back to Part 50.
MEMBER SKILLMAN: Yeah. Right.
MEMBER RAY: Let's make sure we get done here and then maybe we can have some more open discussion. What is a challenging area for sure? I just want to make sure you get through your slides because we do have other stuff we got to do today so -
MS. KAVANAGH: Okay.

MEMBER RAY: -- I don't want to run over.

MR. HEATH: All right. So as I mentioned before, the NRC's QA oversight consists of licensing reviews plus inspections.

One of the findings from NUREG 1055 was the NRC inspection program at the time focused heavily on paperwork. Today, in conjunction with the licensing review, the NRC implements a construction inspection program which consists of a number of direct inspections of the DC COL applicant and their contractors to verify the applicants meeting their requirements and the commitments set forth in the application.

So for the period of time that the DC application is under review in the NRC's inspection manual Chapter 2508, design certification would apply. This inspection manual chapter applies to the applicant and the applicant's contractors and other safety-related activities related to the design and review process.

The purpose of the guidance in this manual chapter is to provide assurance that the application for the design certification meets requirements in subpart B to 10 CFR Part 52.

There are two inspections that we conduct under this inspection manual chapter. The first is to QA program implementation inspection, or inspection procedure 35017.

This is an estimated 200-hour inspection which is conducted at least once and then again as necessary. The purpose is to verify implementation of the QA program as described in the DC application.

The other inspection is the design qualification testing inspection, which is inspection procedure 35034. This is a 240-hour direct inspection which is conducted at least once and then again as necessary and its purpose is to verify whether the qualification testing activities supporting the application are conducted in accordance with Appendix B.

MEMBER RAY: Okay. Now I am going to stop you here and just ask a simple question that hopefully no need for elaboration.

In what you just described, the question has to be raised as to well, given that that was done for the AP 1000 design certification how did it miss what was later deemed to be the cause of the condensate return error?

Has that been looked at? I don't want to try and answer that, your question, but it -- you know, it's the sort of thing that in trying to diagnose whether what we are doing is adequate it's kind of a -- and we participated in it as much as anybody on the staff did.

Not being anybody here other than that clearly it was not something that we would have been expected to catch, we, the agency, other than as the program that was being implemented didn't catch it. That's the issue.

And so the question is how did we fail to recognize that the program that Westinghouse was using to verify things like the assumed condensate return was flawed.

MR. HEATH: So there way -- we may be able to bring a little bit to light in the next inspection that I'll talk about here in a minute.

But I think it's fair to say that the post-docketing QA program inspection is not designed to capture those types of design issues because that's not what we are looking for when you look at the elements -

MEMBER RAY: No, no, no. That's not my point. Let me try one more time and then I'll quit because, again, I don't want to get bogged down here.

The issue is okay, we found an error --

how did the QA program allow that error to occur, and if that's -- if we can identify that, which they did do during the -- during the work that was presented to us a year ago, the next question is given the flaw in the program -- not in the design but in the program -- what does it say about our inspection of that program, which is what you're talking about?

MS. KAVANAGH: Again, our program is just on a sampling basis and I understand. So that error -- that type of error did not show up when we did our inspection. So we didn't identify it during our inspection.

MEMBER BLEY: Can I try something a little different?

MEMBER RAY: Yeah. Absolutely. Of course.

MEMBER BLEY: After that, and we have seen this in another case that we looked at this week where a vendor had a design problem that caused some trouble, the vendor went back and said, our process for looking at these design issues was flawed in some ways and we have beefed up our program in hopes that we won't have this sort of thing happen to us again.

That, I think, is the kind of program that Harold say you look at that kind of program that they have to ensure their design process works well and is there anything more we could do as an agency to help them have a program that's more likely to catch these kind of problems that we have seen.

MR. HEATH: I mean, then again --

MEMBER BLEY: That's what generic lessons learned and we did -- we are good.

MS. KAVANAGH: You know, I understand. But I guess what we are trying -- with our resources and the outcry of regulatory burden we need to step carefully as to what we can and can't do.

MEMBER BLEY: Well, what we are probing here is not -- you do a sampling. You can't look at everything. But shouldn't the overall program be one of the things you're really focused on. Is that thing set up in a way that they are most likely to get a quality product out.

MS. KAVANAGH: Go ahead.

MR. HEATH: So our QA licensing reviews of the program and then these QA program inspections are designed to ensure that the applicant has the necessary Appendix B controls in place to identify these issues.

Now, how those are implemented on behalf of the applicant I can't say NRC has so much control over when they are implementing any individual aspects of that program. And we approved that they have -- we say yea, this applicant has the required regulatory controls in place.

MEMBER RAY: Okay. Look, we have only got 30 minutes so, again, I keep saying this because I don't want to infringe on other things that somebody has to do today.

Let me just say now and we will hopefully move on, it's my experience that when we inspect and find a problem them we ask why did this problem occur.

MS. KAVANAGH: Right.

MEMBER RAY: But we don't put emphasis in the areas that Dennis and I have been talking about, which is do you have in place a program that prevents problems from occurring in the first place, because that's programmatic as opposed to why did this happen.

And I think we are going to want to consider that issue somewhat more. I was hoping that the experience that we had with -- and I'll just keep using comments they return because it's something we all recognize -- we would say uh-huh, we are not adequately ensuring independence of design review when we do these QA program inspections. That's the sort of thing I was hoping we would conclude if it's legitimate to conclude it.

And so we are going to put more emphasis on ensuring criterion 3 allows either design review or testing.

But we are going to put more emphasis on verifying independent design review is occurring because that's something that's expensive and it's readily omitted often.

MR. OH: Mr. Ray, if I can add something.

MEMBER RAY: Sure, and we will move on now but go ahead.

MR. OH: And Jermaine is going to go into in the next couple of slides. But, you know, we are doing engineering design verification inspections which are trying to get into just the areas that you're talking about specific to design activities, kind of lessons learned, what we have seen in some of the results from the component inspectors.

And then we take that knowledge, we go to Westinghouse and we do a specific EDV inspection, and Jermaine will touch on that a little bit.

MEMBER RAY: Okay. But one more time -- I guess I am still of the opinion that we are triggering what we do mostly off of problems that we find in the design rather than looking at the process to see if it conforms with the requirements for independent review. For example, going and say, show me the independent review that was done of this and by whom and how is the results compared.

MR. OH: That is part of our inspection.

MEMBER RAY: Okay. If that's part of it that's what is critical here because we can never find all the problems nor can the vendor unless they apply a program like Appendix B requires.

MR. OH: You're exactly right. When we -- you know, don't get fooled when we say vendor inspection. We are not just looking at the product, which -- don't feel -- we are looking at the implementation of all safety-related activities. We have 18 criteria. We try to inspect all 18. So we'd be looking at procurement, document controls, oversight of supplies, of course. Nonconformance and corrective action -- that's kind of going to be the triggers for are they doing an effective oversight -- are they doing a good design -- independent design verification.

MEMBER RAY: But if the only reason -- the condensate return is just the example to me of is it just because we were sampling? Because to me sampling -- if the problem was the lack of independent review sampling isn't particularly relevant. You either have independent review or you don't.

And with that having been said, though, I don't want to get -- I am getting into debates here that I don't want to do. So let's keep going and see if we can make sure we get to the end because we do have to take any public comments.

MR. HEATH: Okay. So I'll talk a little bit quickly about the vendor inspection program, which is also my portion of the organization, quality assurance vendor inspection branch.

When the DC or COL application is submitted and there are safety-related purchase orders in place, the NRC inspects aspects of the development of the detailed certified design through direct inspection of the applicant's suppliers.

These inspections are accomplished through the vendor inspection program inspection manual Chapter 2507, which establishes the inspection program for vendors who supply safety-related equipment services -- equipment and services to the commercial nuclear industry, and we will implement inspection manual Chapter 2507 for any applicants -- any -- for the applicant's vendors or suppliers referencing the certified design.

The DC applicant is also considered a vendor at this point of the COL applicant. The vendor inspection program is made up of a number of inspections. For completeness I listed them all here but I just want to focus on the last one here, which is the engineering design verification inspection.

So, historically, one of the shortcomings identified in the lessons learned from NUREG 1055 was a failure of the NRC to perform inspections of any depth in the area of design.

So now we have the engineering design verification inspection, which is a large team-based inspection of approximately 1,600 hours and is comprised of a multi-discipline 10 to 15 man or woman team.

The EDV -- engineering design verification -- is a detailed technical review of selected systems. The inspection is also a sampling and it provides the NRC an opportunity to assess the design authority or DC applicant's implementation of its processes for completing and controlling of the detailed design.

The detailed technical review provides reasonable assurance that the design authority's processes are sufficient to result in a complete and accurate transfer of high-level design information contained in the final safety analysis report into detailed engineering procurement and/or construction documents.

The successful completion of this inspection will also provide a reasonable assurance in the validity of the resulting detailed design information which may ultimately be utilized by COL holders to support the closure of ITAAC.

Now, this inspection is conducted when the design is approximately 70 percent complete. That's not a hard number. Normally, we would conduct this inspection when we have enough information for any particular systems which we feel that we could make an adequate assessment to do an evaluation based on the output data that we have.

We also use our probabilistic risk assessments and additional risk insights to try to identify which systems we are going to select for those inspections.

MEMBER REMPE: Before you leave, just to make sure I understand, clearly, you can do design verification aspects of it but are you also considering design development and how the design -- the process used to develop the design such as was an independent review used to develop the design, which is more than just saying going to make a change -- they do things as they are supposed to? And I haven't heard that word develop. You talk about procurement but I haven't heard development of the design. Do you go into that aspect of it?

MR. KROHN: This is Paul Krohn. I am the deputy director for DCIP. I think we could if we were led there, because 1,600 hours lends itself to a programmatic and almost diagnostic type inspection. So could we pull that string? I think we could. Do we commonly? I'd have to defer to the inspectors.

MR. HEATH: I mean, I know we definitely -- a component of that inspection is verifying independent review of the design outputs. I mean, we can easily lead -- lead us into design development based on what we find. But I don't want to speak without have an inspector procedure in front of me how far -- if we are actually looking at development aspects. I mean, I -

MEMBER REMPE: In my opinion, both of the problems that Harold mentioned was a condensate return and what Dennis was referring to was part of how they were developing the design -- what tests were used, if assumptions were questioned. And so maybe that should be considered.

MEMBER RAY: Yeah. I mean, I think to add to that, look, in the condensate return the fundamental problem was an assumption made very early that was never validated until it was challenged across the pond, and then it was found to have been an incorrect and sort of off the cuff assumption.

The -- I am sorry to interrupt but just finish the thought, then I want to get -- if there had been an independent design, the chances of two independent people making the very same assumption could presumably be pretty low and so that would have raised an issue.

But the fact was there wasn't any independent at that level of the design development. Now, there may have been independence in sizing the scuppers that directed the water down the drain and it might be something that we would see here.

But I am really wondering if this inspection that you're describing, Jermaine, is one that would get back that far into where did this assumption come from.

MS. KAVANAGH: Yeah. Typically, this inspection looks at coming from finality forward. Yeah, so we would -- in this particular inspection we would not go back into what we have already approved.

MEMBER RAY: So that's where our concern, I think, probably will remain. I don't know that it's anything that we can do anything about or anybody can do anything about.

But the bottom line is when we are asking these questions that's the sort of thing we are asking about because independence causes something like an out of the air assumption to be revealed as such.

But once you make the assumption the execution of the assumption, my God, you could challenge it from -- until the cows come home and never find a problem. Okay. Please go ahead.

MEMBER CHU: Why is it a QA problem? Why isn't that a technical issue to decide?

MEMBER RAY: QA is not untechnical or nontechnical. QA is simply saying you have two technical people -- two independent technical ways of deciding an issue. That's all QA is doing. It's not substituting process for technical expertise.

It's just saying you need to do it a second time and I can give you lots of in fact very current stuff for me -- areas where there was a lack of independence and the results were a disaster.

MS. KAVANAGH: Right.

MR. HEATH: Well, criterion -- I mean,

criterion 3 it's very clear that that is a requirement -- that an independent verification checking is a requirement.

Now, if there are instances that, you know, those aspects aren't getting implemented, I mean, our review should conclude that the entity has appropriate controls in place and then when we conduct our inspections for the systems that we have sampled we would go to that rigor to look to see how some assumptions -- that someone -- that there was independent verification and those aspects were challenged.

MEMBER RAY: That's right. That's right. And so the issue then becomes did we do that in this case or was it just we did it elsewhere and not here, which is you can't do it 100 percent -- I understand that.

But the -- but the issue is are we checking for independence of design review or are we going forward from the set of developmental assumptions that Joy was referring to and checking from that point forward but not looking at what comes before. Okay. Now, again, I am watching the clock here.

MS. KAVANAGH: I understand, and I am

going to go back to one of these previous slides really quickly. In our -- as part of our DC review -- our application review -- we do conduct at least one inspection where we do go and look at how the design was developed and it's part of the QA.

MEMBER RAY: That's the 200-hour one, yeah.

MS. KAVANAGH: Yeah. It's the QA implementation and it's just of select systems.

MEMBER RAY: And I'll just want to differentiate that, Kerri, from this one.

MS. KAVANAGH: No, I agree. There is a difference.

MEMBER RAY: Okay. Please do proceed and, like I say, we got to get public comments and I got to be done or the chairman will cut my head off.

MR. HEATH: All right. Well, luckily, the next slide is the conclusion. So quality assurance is very integral to the design, fabrication and construction of nuclear power plants.

Proper implementation of Appendix B -- of an Appendix B program focuses on achievement of result, emphasizes the roles, the individuals and management in the achievement of quality and fosters the application of these requirements in a manner consistent with the activities important to safety.

The NRC staff has vigorously reinforced that Appendix B requirement that an applicant retains responsibility for the work delegated to other entities.

The NRC staff concludes that the current QA licensing review and inspection process has proven to be effective. The staff's approval of the applicant's QAPD -- it's the NRC's determination that the applicant's QAPD meets the regulatory requirements of Appendix B.

And while implementation of an Appendix B program does not guarantee that all design issues and construction issues will be identified, an adequate program does provide the detention that provides the reasonable assurance that design and construction issues will be identified properly and controlled.

It is a fact that the AP 1000 design issues that resulted in four of the five exemption requests were identified by DC or the COL applicant, which speaks to the effectiveness of the adequacy of Appendix B controls which allow such design issues to be captured.

Based on the lessons learned from NUREG 10.55, the staff is more engaged in the oversight of the applicant's implementation of his QA program. Upon acceptance of the applicant's QA program the NRC conducts inspections to assess the applicant's translation of its QAPD commitments, its regulatory requirements and design basis information and to procedures, processes and instructions. These inspections focus on the effectiveness of the QAPD implementation.

So the staff will continue to review and improve upon our own guidance including the QA licensing and inspection processes in order to provide reasonable assurance that these system structures and components will be designed and constructed in accordance with the regulatory requirements.

And that would conclude our portion of the presentation.

MEMBER RAY: Okay. So we have got a little less than 15 minutes here during which time we need to see if there are any public comments but also, having gotten to this point, I think there is adequate time for members to ask additional questions if they wish to do so.

MEMBER POWERS: You indicate you use a sampling process to explore what an applicant or a licensee is doing. If I were new to your branch, what would I read to understand how that sampling is done?

MS. KAVANAGH: I would start off with inspection manual Chapter 2502, which talks about our activities in the pre-COL time frame. I would -- I would have you read 2506, which talks about the overall construction inspection program and how all the manuals fit together.

I would have you also read 2508, which talks about our inspections for the DC applicant. And usually it should get into some detail about, you know, we target design areas that the staff has identified as being issues. So we will communicate with our other technical staff to find out what are the issues you're having -- where would you like us to spend our time and focus on.

MEMBER POWERS: Having gotten my feet wet reading the theory, what example QA plan would you recommend I read in order to understand how the theory gets applied in a practical case.

MS. KAVANAGH: The QA program that our SCRs do not get into the level of detail where you would see implementation. You might want to start off with NEI templates.

There is NEI 0614 or NEI 1104 where we have approved what a QA program template would look like. But we'd actually have to get you out into the field and have you start polling implementing procedures and how they provide the how-tos to meet the regulation.

MEMBER POWERS: You don't have a document that says okay, for this particular licensee here's what I am going to go and -- here, team, this is what we are going to inspect?

MS. KAVANAGH: No, we do not.

MEMBER POWERS: Just want to -

MR. HEATH: So that's -- I mean, that's easier to do for -- I mean, I would say the process is more designed for Part 50 -- Part 50 plants.

There is --

MEMBER POWERS: Yeah. I know what you mean -- what you're saying, yeah.

MR. MCINTYRE: And as always, standard view plan 17.5, which is our guideline that we evaluate the QA program submittals against so if you look at -- you know, if you look at, say, NEI 1101 and 17.5 you have a pretty good idea of what a QA program submittal should look like.

MEMBER BLEY: For what Dana raised, would any of the inspection procedures be worth looking at?

MS. KAVANAGH: Absolutely.

MEMBER BROWN: Yeah, I just wanted to springboard off of Joy's comments and some of yours and what I've heard in some of the other presentations we have had, not just directly.

But it -- every time we seem to talk about the QA programs it's almost as if the primary focus is on -- totally on process and not -- and when you talk about sampling, the sampling has almost been -- and I am not saying -- I don't -- just taken negatively almost kind of random you pick stuff and do it, and I've never heard where you go back and you look during the design cert phase when you're doing your design inspections, where you look for critical -- the technical -

MEMBER POWERS: That's what she said. They look at the SCR and talk to their staff about the --

MR. HEATH: Are you talking about inspections or are you talking about licensing reviews?

MEMBER BROWN: I am trying to get -- I am trying to get -- well, for example, when I've asked questions in my area relative to what's called a secure development operational environment for the I&C world, it's all process. There is no -- there is no technical aspect that say how do you make sure software's okay -- how do you make sure it's -- and that's appropriate in many areas. I've always looked at the AP 1000 issue that continues to refunction here and this is fundamentally an assumption here. It was in a technical area. It was a critical area in terms of making this plant work properly and it was almost like why wasn't -- we are at fault also maybe because we were buried in this thing as well.

Why didn't we ask that question about where is that technical oversight or in the questions where you say hey, this is a critical area -- what are the assumptions that we should try to make sure before they get too far in the design that this is really correct. Whether it's an independent set of folks or even a more questioning part on that. I don't --

MEMBER RAY: Interestingly, remember Syed did it on the outside of the containment and what we needed was somebody to do it on the inside as well.

MS. KAVANAGH: I mean, we can show you some of our successes as part of vendor inspections. We -- if that's what you'd like to hear.

MEMBER RAY: Oh, no. No. I am just -- and no, I am not -- I am not sitting here trying to be critical on that. It's a matter of when you establish these QA programs to go back and look at these things don't think necessarily in terms of process but what's the process -- where do I focus my process -- what are the major technical issues.

A major one in one of the designs -- you know, it's like the, what, it's the dump valves where they have exploding, you know, actuating the -

MS. KAVANAGH: The squib valves.

MEMBER RAY: The squib valves. Thank you very much. That's one of my --

MS. KAVANAGH: We were -- we were very heavily involved with this.

MEMBER RAY: Yeah, I know. I am -- don't even get me started on squib valves. Okay.

MEMBER POWERS: Well, I came away with the impression that what you told me was that you look at the SCR and talk to the staff about these designs. You identify where the SCR folks thought that there were significant issues. That's the note I wrote down.

MS. KAVANAGH: That is a true statement. But we -

MEMBER POWERS: So, I mean, it seems to me that it's not -

MS. KAVANAGH: But we also --

MEMBER POWERS: -- it's not random. It's not -

MEMBER BROWN: I am just giving -- I am just giving you math labor of participating in a number of these new reactor design process. I'll stop right there. I don't want Harold to take my head off before Dennis takes his off.

MEMBER SKILLMAN: Can I just ask one quick question, please, to Jermaine and to Kerri? Thank you. But let me ask this.

Jermaine, in your third bulletin, slide 12, you've combined DC and COL in a sentence. Let me ask you this. Is it more accurate to communicate that a DC applicant retains responsibility for the establishment and execution of the QA program for the DC material but NRC provides oversight of its implementation and independently the COL applicant retains responsibility for the establishment and execution of the QA program under its COL application?

MS. KAVANAGH: That's a correct statement.

MEMBER SKILLMAN: Okay. Now, hold that thought. If you were to present that information with that separation then I am wholeheartedly behind you. Once those two become combined, in my view, a gray area appears, because each can go like this.

MS. KAVANAGH: Right.

MEMBER SKILLMAN: In my view, the DC applicant has a unique responsibility for the thoroughness and adequacy of its information under its Appendix B program and the organization that buys that design or adopts that design because it's now owned by the public has the accountability to proceed with that DC information and incorporate it into its COL under its own QA program.

MS. KAVANAGH: That's correct.

MEMBER SKILLMAN: If you agree with that, then I am 100 percent behind you.

MR. HEATH: That is correct. But under Part 50 you still get some of --

MEMBER SKILLMAN: Got that. I understand that.

MR. HEATH: Well --

MEMBER SKILLMAN: But I am trying -- what I am really trying to do is to make sure that the burden for accuracy, say, for the Westinghouse condensate stuff, really rests with the NSSS designer, not with a portion in North Carolina who's trying to implement it because that entity does not have the codes, the standards, the DECs, the reams and reams of documentation, all the Criterion 3 stuff that was embedded in the DC. That poor individual -- that poor entity comes in and says, I want to -- I want to build that one.

MEMBER RAY: Okay. That's an individual member comment -

MEMBER SKILLMAN: It is.

MEMBER RAY: -- we could debate for some time but we won't. We should ask if there are any public comments. Yes, sir.

MEMBER BLEY: Before you do, I have a question. It's kind of a where do we go from here question. This has given us a lot of information. Are we looking to follow this or are we going to follow it the way you started with the meeting we already have scheduled and this is background information?

MEMBER RAY: I recommend the latter because we already are going to be spending time deciding whether we want to engage with this Reg. Guide that you saw listed up there.

I think we ought to mull it over and discuss it at P&P tomorrow, Dennis, because members may come to a different conclusion than this. We may feel like we have got enough information to make a comment just based on what we have heard today. But

I am skeptical if that's the case. I am thinking we would want more information than we have now before making a comment. But I could change my mind about that.

MEMBER BLEY: I think you can go to public comment.

MEMBER RAY: Yes, sir. Okay. With that then having been said, we will ask to -- anyone on the bridge line, and there was someone on when we began this meeting, who would like to make a comment please do so.

MR. BROWN: Bridge open.

MEMBER RAY: Yes, please.

MEMBER BLEY: He said the bridge is open.

MEMBER RAY: Bridge is open. Thank you. If there is anyone please do go ahead. Is there anyone in the audience who would like to make a comment?

Well, that's good. I want to express my appreciation to staff for their response here and particularly for the diligent way in which they tracked the very specific questions that we had to try and help get some focus on what we are talking about here. We have had some comments by members, not comments from the committee but comments from members, myself included, that you would recognize as not fully satisfied by what we have heard today. But this is a very difficult area, in my judgment, for reasons that Member Rempe mentioned because we are just looking ahead. We are not trying to look back on anything except when that allows us to learn something that's appropriate for the future.

And whether there is anything that needs to be further -- I mean, I do believe that the experience that led us to this point -- the AP 1000 experience -- is doubtless a motivation in your organization to do what needs to be done.

But, you know, that can only last so long and the question is, is there a -- is there something else that we need to do including acknowledging if there is a progressive certification process in the future. Some are limited satisfaction of Appendix B for those early reviews, not to diminish their value but to say listen, we are not done with the quality program issues. And I think the best example to use is the one we keep coming back to.

It made an assumption -- it was the wrong assumption but nobody questioned it for years until, like I said, it got across the pond. Somebody asked the question. But there was also testing done on the sheathing of the water sheeting on the outside of the containment.

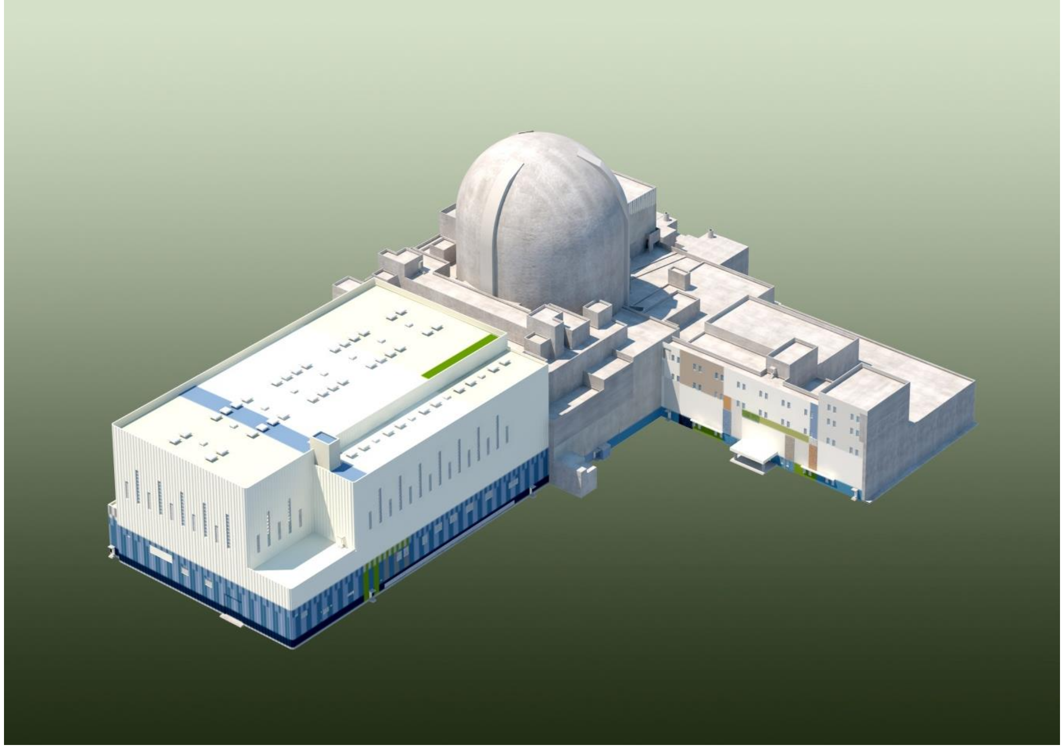
That question did get asked in this room. The answer was oops, we made a mistake and it got fixed. But we can't rely on that for late problems. They are issues that we hope never will even arise, much less reveal deficiencies.

With that, I've spent my time and I turn it back to you, Mr. Chairman.

MEMBER BLEY: Thank you, Mr. Ray. We are at this point off the record.

(Whereupon, the above-entitled matter went off the record at 4:14 p.m.)

APR1400 DCA
Chapters 2, 5, 8, 10, 11



ACRS Full Committee Presentation

KEPCO/KHNP

February 9, 2017

Introduction

- **The APR1400 is essentially complete design**
  - The Shin-Kori Unit 3 went into commercial operation (Dec. 20, 2016)
  - Baraka NPP unit 1-4 under construction in UAE
- **Distinguished design features of APR1400**
  - Advanced Accumulator – fluidic device in safety injection tank
  - Overpressure protection – POSRV
  - RMI to address GSI-191, Seismic design with FEM model
  - Enhanced SBO Coping Capability - gas turbine generator for AAC source, 16 hr battery(Train C/D)
  - Improved tolerance to the beyond design basis
    - Aircraft impact analysis by 10CFR50.150
    - LOLA(Loss of Large Area) and Physical security design

Chapter 2: Site Characteristics

- **The site interface requirements for APR1400 design include geological, seismological, hydrological and meteorological characteristics.**

- **The APR1400 is designed on the basis of a set of assumed site-related parameters. (DCD Chapter 2 Table 2.0-1)**

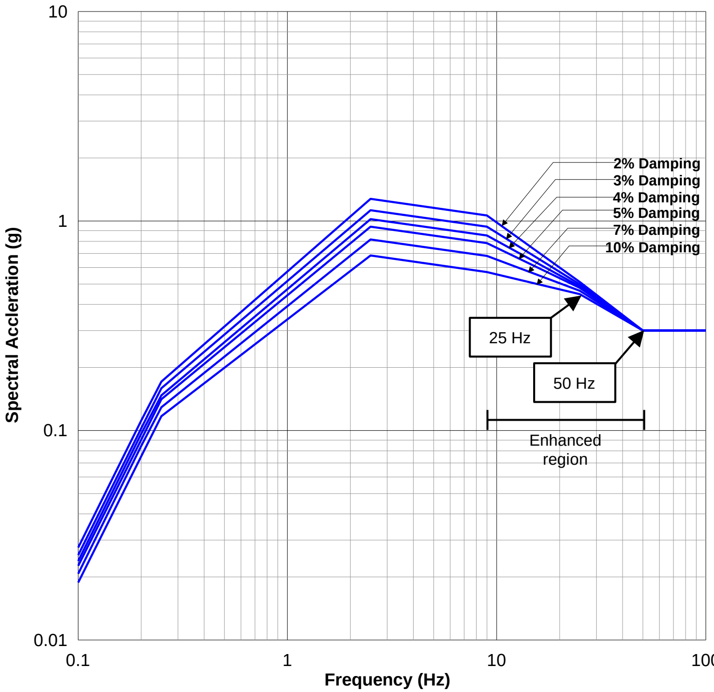
- **The COLA is to confirm site characteristics are bounded, or provide site specific qualification.**

Chapter 2: Site Characteristics

- **Atmospheric dispersion factors**
  - Bounding atmospheric dispersion factors and deposition factors were used for dose evaluations
    - Long-term χ/Q & D/Q : the bounding values of 2.0E-05 s/m^3 for χ/Q and 2.0E-07 m^{-2} for D/Q at EAB
    - Short-term χ/Q : 1.0E-03 s/m^3 at EAB based on EPRI-URD
  - The 95th percentile on-site atmospheric dispersion factors were used for the MCR/TSC habitability analyses
    - The most limiting meteorological condition were selected among the 6 U.S. NPP sites and the 50% margin was added in the on-site χ/Q evaluation results to envelope most of the U.S. site meteorological conditions

Chapter 2: Site Characteristics

- **The peak ground acceleration of SSE(Safe Shutdown Earthquake) is 0.3g for the APR1400 standard design.**
- **The CSDRS (Certified Seismic Design Response Spectra) are used as the design response spectra for the SSE.**



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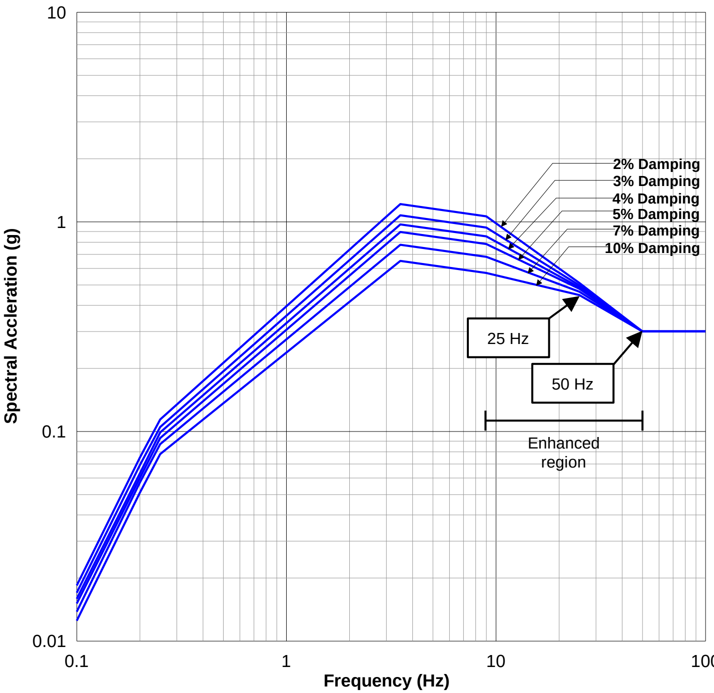

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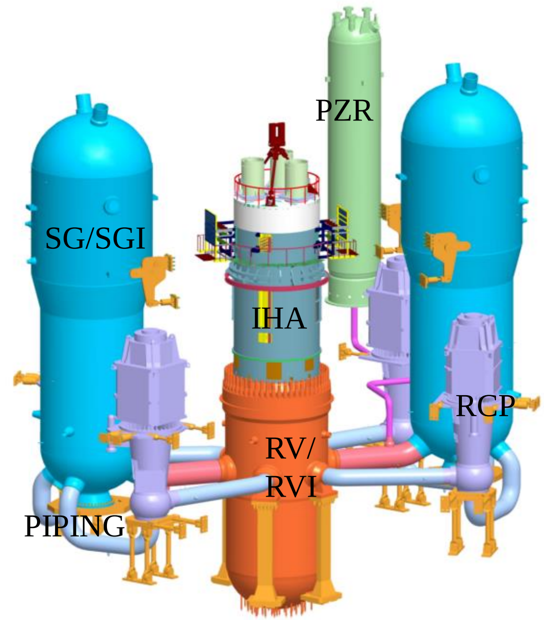
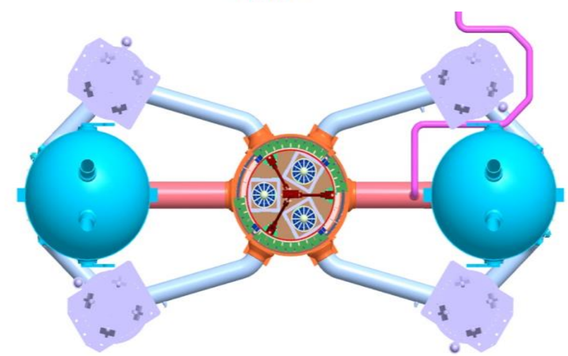
- **There is no technical issue in Chapter 2.**

Chapter 5: Reactor Coolant System and Connecting Systems

- **Design features of reactor coolant system**

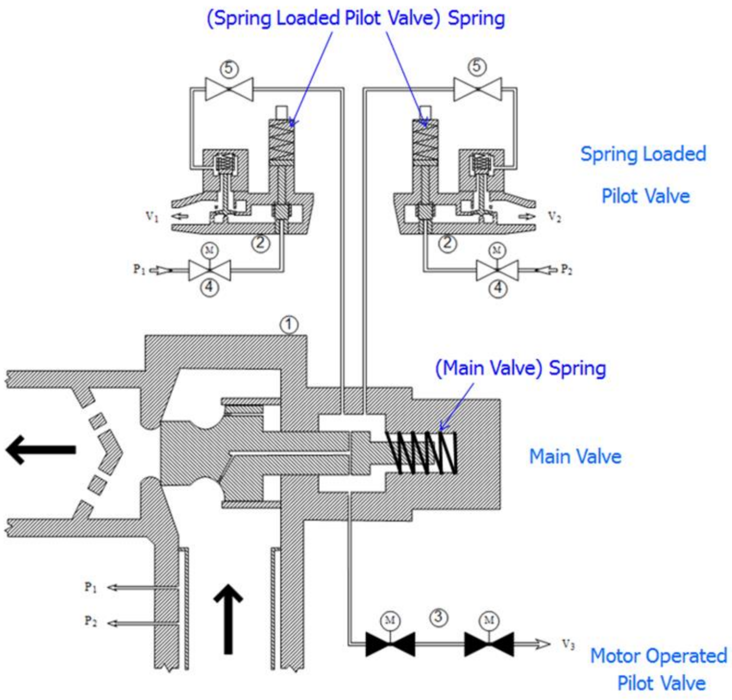
  - Design Life: 60 Years
  - Power: 4,000 MWth / 1,400 MWe
  - Two Loops: Symmetric
    - ✓ 1 RV, 2 SGs, 4 RCPs, 1 Pressurizer
    - ✓ 2 Hot Legs, 4 Cold Legs
  - Primary Operating Condition:
    - ✓ Pressure: 2,250 psia
    - ✓ NOP Hot / Cold Temp.: 615 / 555 °F
  - Secondary Operating Condition:
    - ✓ Pressure: 1,000 psia
    - ✓ MF/MS Temp.: 450 / 545 °F
  - SG Tube: 13,102 /SG, Alloy 690TT
  - Pressurizer with POSRVs





Chapter 5: Reactor Coolant System and Connecting Systems

- **Overpressure protection**
  - Pressurizer POSRV
    - Main Valve (1)
    - Spring Loaded Pilot Valve (SLPV) (2)
      - ✓ Automatic actuation for RCS overpressure protection
    - Motor Operated Pilot Valve (2)
      - ✓ Manual actuation for RCS rapid depressurization



  - Low Temperature Overpressure Protection (LTOP)
    - Shutdown cooling system (SCS) suction line relief valve
  - MSSVs for secondary overpressure protection

Chapter 5: Reactor Coolant System and Connecting Systems

- **Material design for RCS components**
  - Proven materials through successful plant operating experiences and Code & Licensing requirements
  - Resistant to reactor coolant environments and various degradation mechanisms such as;
    - Corrosion, Stress Corrosion Cracking, Fatigue, Radiation effects
  - A minimum number of types and grades of materials for RCS components
    - Low-alloy steels with austenitic stainless steel or nickel-based alloy cladding for RCS Components, RCL Piping, and large Nozzles
    - Alloy 690TT for SG Tubes, CEDM, ICI, and small Nozzles and Alloy 690 equivalent materials for their weld metals
    - Austenitic stainless steels for rector internals and core support structures

Chapter 8: Electric Power System

- **4 Train Design - Class 1E Onsite AC&DC Power System**
  - Each redundant load group (division) consists of a divisional pair of trains: Trains A plus C for Division I; Trains B plus D for Div. II.
  - Dedicated Class 1E EDG for each train
    - Trains A & B: 9100 kW
    - Trains C & D: 7500 kW
  - Each train is physically and electrically independent of non-Class 1E power system and other Class 1E trains.
  - No interconnection and load share between trains.
  - Equipment is located in dedicated locations in accordance with quadrant division arrangement.

Chapter 8: Electric Power System

- **Enhanced SBO Coping Capability**
  - 16 hour SBO coping duration per RG 1.155
  - Diversified AAC source (i.e., gas turbine generator in lieu of diesel generator) from the emergency AC sources.
  - The AAC GTG will be aligned to a shutdown bus (train A or B) within 10 minutes of the onset of an SBO.
  - The AAC GTG has sufficient capacity (9,700 kW), capability, and reliability (0.95) to bring the plant into safe shutdown condition (i.e., hot shutdown condition).
  - 16 hour duty cycle (without load shedding) is considered for DC trains C and D batteries to support trains C and D plant equipment (e.g., TDAFWP) during an SBO.

Chapter 8: Electric Power System

Open items

- Open Phase Conditions (OPCs)
  - KHNP provided a response to RAI 8521 as of Nov. 14, 2016, which includes the design vulnerability study and a set of DCD mark-ups that incorporated the design features of OPD system, necessary COL items, and ITAAC.
- Design Compliance with SECY-91-078
  - KHNP provided a response to RAI 8426 as of Jan. 4, 2017, which includes appropriate explanation and justification as to how the APR1400 offsite power system design satisfies the GDC 17 and SECY 91-078 requirements in detail manner.

Chapter 10: Steam and Power Conversion System

- **Converts the heat generated by the reactor into electrical energy by using condensing cycle with regenerative feedwater heaters.**
- **T/G system**
  - 1,800 rpm turbine, two sets of MSRs, generator, exciter, controls, and associated subsystems.
  - double-flow high-pressure (HP) turbine, three double-flow low pressure(LP) turbines, and a direct-coupled generator in tandem.

Chapter 10: Steam and Power Conversion System

- **Main steam system**
  - Four main steam lines with Two SGs
  - Each steam line has MSADV(1), MSSVs(5), MSIV(1), and MSIVBV(1)
  - Main steam line supplies the steam to the turbine-driven auxiliary feedwater pump.
  - The turbine bypass system has the capacity to bypass 55% of the rated steam flow to the condenser with eight (8) TBVs.
- **Auxiliary feedwater system**
  - Each division has 100% motor-driven AFWP(1), 100% turbine-driven AFWP(1), and 100% AFWST(1)
- **Piping materials of steam and feedwater system**
  - Cr-Mo Alloy steel to prevent the FAC (Flow-Accelerated Corrosion)
  - Carbon steel pipe adopts additional thickness
- **No issues to be discussed in Chapter 10**

Chapter 11: Radioactive Waste Management

- **Source terms**
  - Design Basis Source Terms
  - Expected Source Terms
  - Secondary System Activity
  - Source Terms for Radwaste Systems
- **Radwaste management systems**
  - Liquid Waste Management System
    - R/O technology / Pretreatment to remove the organic / Ion exchangers
  - Gaseous Waste Management System
    - Uses charcoal delay beds to delay Xe for not less than 45 days
  - Solid Waste Management System
    - Uses spent resin drying system / spent resin long term storage tanks / Solid waste compactor / Filter handling system

Chapter 11: Radioactive Waste Management

- **Process & Effluent Radiation Monitoring and Sampling System (PERMSS)**
  - Measures and records radioactivity levels of the liquid and gaseous process streams and effluents from LWMS, GWMS and other process systems during normal operation, AOO, and Postulated Accident.
  - Following PERMSS are used
    - Gaseous PERMSS monitors (GWMS, HVAC)
    - Containment air monitor and MCR air intake monitor.
    - Main steam line monitor (N-16)
    - Liquid PERMSS monitors (LWMS, CVCS, CPP, CCW)
    - Steam generator blowdown monitor
    - Essential service water pump discharge monitor
    - Fire pump and waste water treatment building drain monitor
    - Condenser pit sump water monitor
    - Condensate receiver tank monitor

Chapter 11: Radioactive Waste Management

- **Offsite public doses due to normal operation**
  - Estimated doses to max. organ are 14.5 mrem/yr(gas) and 4.05 mrem/yr(liquid)
  - Design basis effluent concentrations at EAB are within 18.0% of ECL for liquid and 16.2% for gaseous effluents
- **Radwaste system failure analysis**
  - Liquid radwaste system failure analysis
    - Boric acid storage tank (BAST) determined to cause the worst consequence
    - Minimum required dilution factor which can meet the concentration limits for potable water in 10 CFR 20 App.B was estimated to be 9,340
  - Gaseous radwaste system failure analysis
    - Inadvertent bypass of charcoal delay beds in GWMS is assumed
    - Estimated doses at EAB and LPZ are 1.16 mrem, 0.255 mrem
- **There are currently three Open Items associated with Chapter 11 that are under NRC evaluation.**

Attachment : Acronyms

AAC	*Alternative AC*
AFWP	*Auxiliary Feed Water Pump*
AOO	*Anticipated Operational Occurrence*
CEDM	*Control Element Derive Mechanism*
EAB	*Exclusive Area Boundary*
FEM	*Finite Element Method*
GTG	*Gas Turbine Generator*
GWMS	Gaseous *Waste Monitoring System*
ICI	*In Core Indicator*
LWMS	*Liquid Waste Monitoring System*
MSADV	*Main Steam Atmospheric Dump Valve*
MSR	*Moisture Separating Reheater*
MSIV	*Main Steam Isolation Valve*
MSIVBV	*MSIV Bypass Valve*
MSSV	*Main Steam Safety Valve*
OPD	*Open Phase Detection*
POSRV	*Pilot Operated Safety Relief Valve*
RMI	*Reflective Metallic Insulation*
RCS	*Reactor Coolant System*
RCL	*Reactor Coolant Loop*
SBO	*Station Block Out*
TDAFWP	*Turbine Driven Aux. Feed Water Pump*
TT	*Thermal Treated*
T/G	*Turbine Generator*

Presentation to the ACRS Full Committee – 640th Meeting

Advanced Power Reactor 1400 (APR1400) Standard Design Certification

Safety Evaluation Report with Open Items for Chapters 2, 5, 8, 10, & 11

Tarun Roy, Jessica Umana, George Wunder
APR1400 Design Certification Project Managers

February 9, 2017

APR1400 Design Certification Review Schedule

	COMPLETION DATE
Phase 1 – Preliminary Safety Evaluation Report (SER)	Completed
Phase 2 – SER with Open Items	March 2017
Phase 3 – ACRS Review of SER with Open Items	June 2017
Phase 4 – Advanced SER with No Open Items	December 2017
Phase 5 – ACRS Review of Advanced SER with No Open Items	June 2018
Phase 6 – Final SER with No Open Items	September 2018
Rulemaking	May 2019

Summary of the APR1400 Safety Evaluation Reports

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- The staff has issued Safety Evaluation Reports (SERs) with Open Items for Chapters 2, 4, 5, 8, 10, and 11.
- Of the issued chapters, all but Chapter 4 have been presented to the Advanced Power Reactor 1400 (APR1400) ACRS Full Committee.
- The staff has also issued SERs with no issues for 3 APR1400 Topical Reports.

APR1400 Chapter 2
Site Characteristics

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Technical Topics of Interest

- As the following information is site specific, the combined operating license (COL) applicant is required to provide this site specific information.

Section 2.1 - Geography and Demography

- The COL applicant is to provide this site specific information as part of COL information Item 2.1(1) in the COL application.

Section 2.2 - Nearby Industrial, Transportation, and Military Facilities

- The COL applicant is to provide this site specific information as part of COL information Item 2.2(1) and COL information Item 2.2(2) in the COL application.

Section 2.3 - Meteorology

The SER for Section 2.3 addresses:

- Regional Climatology
- Local Meteorology
- Onsite Meteorological Measurements Program
- Short-term Atmospheric Dispersion Estimates for Accident Releases
- Long-term Atmospheric Dispersion Estimates for Routine Releases.

- Staff reviewed the adequacy of the DCD site parameters related to Regional Climatology, Short-term Atmospheric Dispersion estimates, and Long-term Atmospheric Dispersion and Deposition estimates.
- The COL applicant is to perform the radiological consequences analysis to demonstrate that the related dose limits specified in 10 CFR 50.34 and GDC 19 are not exceeded, if the site-specific χ/Q values exceed the bounding values described in Table 2.3-1 to 2.3-12 of the FSAR.
- All regulatory requirements for Section 2.3 have been satisfied.

Section 2.4 – Hydrologic Engineering

The SER for Section 2.4 addresses:

- Hydrological description
  - Floods
  - Probable Maximum Flood on Streams and Rivers
  - Potential Dam Failures
  - Probable Maximum Surge and Seiche Flooding
  - Probable Maximum Tsunami Flooding
  - Ice Effects
  - Cooling Water Channels and Reservoirs
  - Channel diversion
  - Flooding Protection Requirements
  - Low Water Considerations
  - Groundwater
  - Accidental Release of Liquid Effluents in Ground and Surface Water
  - Technical Specifications and Emergency Operations Requirements

- All regulatory requirements for Section 2.4 have been satisfied.

Section 2.5 – Geology, Seismology, and Geotechnical Engineering

The SER for Section 2.5 addresses

- Geologic
- seismologic
- Geotechnical site parameters used for APR 1400 structural design and analysis

  - Applicant properly specified appropriate geologic, seismologic and geotechnical site parameters

  - All regulatory requirements for Section 2.5 have been satisfied.

APR1400 Chapter 5
Reactor Coolant System and Related Systems

- The staff's areas of review for Chapter 5 covered the reactor coolant system including the reactor vessel, steam generators, reactor coolant pumps, pressurizer, and associated piping

- Most of the regulatory requirements for Chapter 5 have been satisfied.

- The remaining issues
  - Shutdown Cooling
  - Reactor Coolant Pressure Boundary
  - Reactor Coolant Pump Flywheel Integrity

APR1400 Chapter 8
Electric Power System

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- The SE for Chapter 8 addresses the offsite power system, the onsite AC and DC power systems, and station blackout
- With the exception of two issues all regulatory requirements have been met.
- There were five Chapter 8 unresolved items.
  - These items all stemmed from three questions on two technical issues
  - Four of the items related to demonstrating conformance to SECY-91-078 as a means of showing compliance with GDC 17
  - One of the items related to addressing the open phase issue raised in Bulletin 2012-01.

APR1400 Chapter 8 Conformance to SECY-91-078

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- The Commission approved SECY 91-078 to give guidance on acceptable means of meeting various regulatory criteria.
  - Guidance states that there should be at least one offsite circuit to each redundant Class 1E (safety) division should be supplied directly from one of the offsite power sources with no intervening non-Class 1E (non safety-related) buses in such a manner that the offsite source can power the safety buses if any non-safety bus should fail.

- The staff requested the applicant to address the following concern:
  - The electric power distribution system includes unit auxiliary transformers (UATs) and standby auxiliary transformers (SATs) with a 4.16 kV common transformer winding feeding both safety and non-safety systems.
  - There is potential for the safety systems to be impacted because of failure in the non-safety systems, and the analysis had not been done to support the proposed design changes of two circuit breakers in series.

APR1400 Chapter 8 SECY-91-078 (Continued)

- Connection of safety bus offsite power sources through non-safety buses in Response to RAI 8426
  - The APR1400 does not have an intervening non-safety bus in the current offsite to onsite electrical configuration; however, the design does include non-safety and safety buses coming from the same secondary side 4.16 kV transformer winding.
  - The applicant provided a failure mode effects analysis (FMEA) to demonstrate that a failure of a non-safety bus or connection will not impact the safety bus.

- The staff's concerns associated with feeding both safety and non-safety loads from the same transformer winding include (1) voltage regulation of the safety buses, (2) transients caused by non-safety loads impacting the safety buses, and (3) failure points between the offsite power supply and the safety buses.

APR1400 Chapter 8 SECY-91-078 (Continued)

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- Voltage regulation of the safety buses
  - The on-load tap changers at the primary side of the UATs and SATs ensure that the medium voltage safety buses are maintained in an acceptable range.
- Transients caused by non-safety loads impacting the safety buses
  - Transients such as motor starting, motor re-acceleration during a bus transfer, and short circuit on a non-safety bus were assessed and the studies showed that the safety systems would be able to perform their intended function.
- Failure points between the offsite power supply and the safety buses.
  - An electrical fault (short circuit fault or ground fault) on a connection to safety or non-safety bus will be detected by UAT (or SAT) relays and allows transfer of power to the alternate PPS or to the EDG power source.
- The staff finds that the applicant’s response is acceptable because it addressed the staff’s concerns discussed above, and provides a FMEA which demonstrates that the APR1400 offsite power system retains its ability to power the safety loads upon a failure of the non-safety bus.
- The staff will request in an RAI that the applicant document in the DCD how transients on the non-safety bus will not impact the safety bus.

APR1400 Chapter 8 Open Phase Condition

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- **Open Phase Conditions**– Requested that the applicant explain how its electrical system design would detect, alarm, and respond to a open phase conditions, with/without a high impedance ground
  - To meet the GDC 17, the applicant should describe how its electrical system design would detect, alarm, and respond to open phase conditions.
  - Staff has received the follow-up RAI response dated November 14, 2016.
  - Staff finds that the applicant’s open phase detection (OPD) system on the primary side of the MT and SATs conforms to the BTP 8-9 for detection of open phase conditions and alarm in the main control room.
  - Staff intends to issue RAI 8729 regarding the protection features for open phase conditions per BTP 8-9. Specifically, the staff will request information on how the failure of the non-Class 1E scheme (i.e. failure of the OPD system, which is the failure of the redundant detection subsystems) does not preclude the onsite electrical power system from performing its safety function given a single failure in the Class 1E onsite system. Furthermore, staff will request information on how the protective actions to automatically protect the Class 1E system against OPC are in accordance with IEEE Std. 603-1991 and 10 CFR 5055a(h)(3).

APR1400 Chapter 10
Steam and Power Conversion System

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- The SE for Chapter 10 addresses the turbine generator (TG), main and auxiliary steam systems, main and auxiliary feedwater systems, condensers, circulating water and steam generator blowdown.
- With the exception of five items all applicable regulatory criteria have been met
- There were nineteen Chapter 10 unresolved items
  - Four items were associated with the turbine generator speed control and overspeed protection issue.
  - One item was associated with auxiliary feedwater system reliability
  - The remaining items asked for clarification, additional detail, or reconciliation of apparent inconsistencies. These are all either confirmatory or easily resolved.

APR1400 Chapter 10 Turbine Generator Overspeed Protection

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- The discussion of Section 10.2 focused on TG overspeed protection
  - Overview of Overspeed Protection System
    - Normal TG Control system
    - Mechanical overspeed protection system
    - Electrical emergency trip system
    - Mechanical and electrical provide diversity
  - COL Applicant to provide design details so that material in DCD meets the definition of essentially complete design.

APR1400 Chapter 11 Radioactive Waste Management

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- The SER for Chapter 11 addresses the Source Terms, Liquid Waste Management System (LWMS), Gaseous Waste Management System (GWMS), Solid Waste Management System (SWMS), and the Process and Effluent Radiation Monitoring and Sampling System (PERMSS).
- The 3 open items remain to be resolved and were discussed during the October subcommittee meeting:
  - Seeking DCD updates for the liquid effluent tracking process for detergent radwaste system.
  - Request for additional information on the descriptions provided for the GWMS Radiation Monitors and LWMS Radiation Monitors 2 questions.

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Presentation to the ACRS Full Committee – 640th Meeting

Advanced Power Reactor 1400 (APR1400) Topical Report

Safety Evaluation Report
for
Fluidic Device Design for the APR1400
(APR1400-Z-M-TR-12003)

Jessica Umana
APR1400 Topical Report Project Manager

February 9, 2017

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- **Project Managers**
  - Jessica Umaña – Project Manager
  - Jeff Ciocco – Lead Project Manager
- **Technical Staff**
  - Matt Thomas – Reactor Systems Reviewer

APR1400 Topical Report
Fluidic Device Design for the APR1400 Review Areas

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- Overall design concept and operation
- Full scale tests (VAPER Test Facility)
- Dissolved nitrogen effect
- Uncertainty analysis

APR1400 Topical Report
Fluidic Device Design for the APR1400
Overview of Safety Evaluation

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- In this safety evaluation:
  - Staff approved the applicant’s development of the SIT-FD
  - Staff approved the applicant’s full-scale testing results meet the applicant’s specific set of design criteria
  - Staff did not approve that the applicant’s specific set of design criteria meet GDC 35 nor 10 CFR 50.46

APR1400 Topical Report

Fluidic Device Design for the APR1400 Issues

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- Effect of smaller break sizes on SIT-FD performance

- Effects due to cavitation

APR1400 Topical Report
Fluidic Device Design for the APR1400
Conclusion

- Full scale test facility provides a sufficient and adequate means for testing the SIT-FD to validate the performance of it against the APR1400 design requirements
- Full scale tests demonstrate and confirm SIT-FD's passive flow control
- The performance and design of the SIT-FD tested in the VAPER facility satisfies the design requirements of the APR1400 SIT-FD
- Manufacturing tolerances and dissolved nitrogen have insignificant effect on observed pressure loss coefficient
- The design requirements of the APR1400 bound all full scale experimental results with uncertainties
- Topical Report acceptable

Regulatory Guidance for Evaluating the Effects of Light Water Reactor Water Environments in Fatigue Analyses of Metal Components

(Proposed Revision 1 to Regulatory Guide 1.207)

Rob Tregoning
Nuclear Regulatory Commission

Gary Stevens
Structural Integrity Associates

Omesh Chopra
Argonne National Laboratory (retired)

Meeting of Advisory Committee on Reactor Safeguards
Thursday, February 9, 2017
NRC Headquarters
Rockville, MD

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Issue Summary

United States Nuclear Regulatory Commission
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- Revising guidance for environmentally assisted fatigue (EAF)
  - Regulatory Guide (RG)
    - Draft Regulatory Guide DG-1309, "Guidelines for Evaluating the Effects of Light Water Reactor Coolant Environments in Fatigue Analyses of Metal Components"
  - Supporting technical basis
    - Draft NUREG/CR-6909, Revision 1, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials"
- Briefed ACRS - Metallurgy and Reactor Fuel Subcommittee (December 2014)
- Released both draft documents for public comment (2014 – 2015)
- Received public comments on both documents (2014 – 2015)
- Developed responses to comments (2015 – 2016)
- Modified documents as a result of public comments (2015 – 2016)
- Briefed ACRS - Metallurgy and Reactor Fuel Subcommittee (December 2016)
- Soliciting ACRS support for final regulatory guidance (February 2017)
- Planning to finalize regulatory guidance for EAF in support of subsequent license renewal (March 2017)

Meeting Agenda

- Background
  - Environmentally Assisted Fatigue (EAF)
  - Current NRC guidance on EAF
  - Proposed revision of NRC guidance
- NUREG/CR-6909, Revision 1
  - Overview of public comments
  - Sample public comments and responses
  - Changes to document
- Revision 1 of Reg. Guide 1.207
  - Overview of public comments
  - Sample public comments and responses
  - Changes to document
- Current status and next steps

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Background:
Environmentally Assisted Fatigue (EAF)

Cumulative Usage Factor (CUF)

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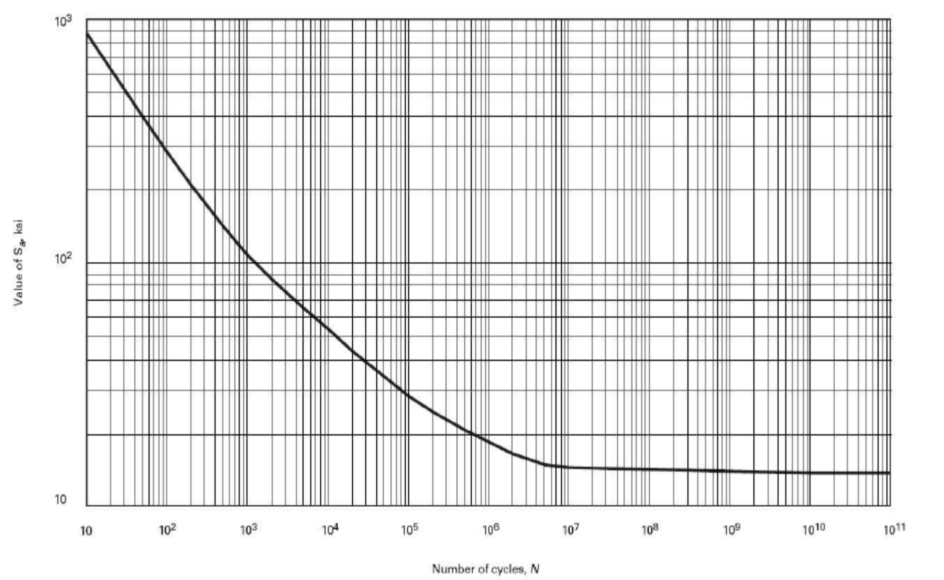
- For nuclear plant design, cumulative fatigue damage due to applied cyclic loading is estimated using cumulative usage factor (CUF)

$$\mathbf{CUF} = \sum_{i}^{Z} \frac{n}{N} = U_1 + U_2 + U_3 + \ldots + U_Z < 1.0$$

where: n *is the applied number of cycles for load* i
N *is the allowable number of cycles for the stress associated with load* i
Z *is the number of applied loads*

- N is a function of the alternating stress, S_a, applied to a component, and is material dependent
- S-N design curves ("fatigue curves") are provided in ASME Code, Section III, Mandatory Appendix I for different materials
- ASME fatigue curves are based on best fits of air test data with design factors applied to account for aspects such as data scatter, size effect, surface finish, atmosphere



GENERAL NOTES:
(a) $E = 28.3 \times 10^6$ psi
(b) Table I-9.2 contains tabulated values and a formula for an accurate interpolation of this curve.

FIG. I-9.2M DESIGN FATIGUE CURVES FOR AUSTENITIC STEELS, NICKEL–CHROMIUM–IRON ALLOY, NICKEL–IRON–CHROMIUM ALLOY, AND NICKEL–COPPER ALLOY FOR TEMPERATURES NOT EXCEEDING 425°C

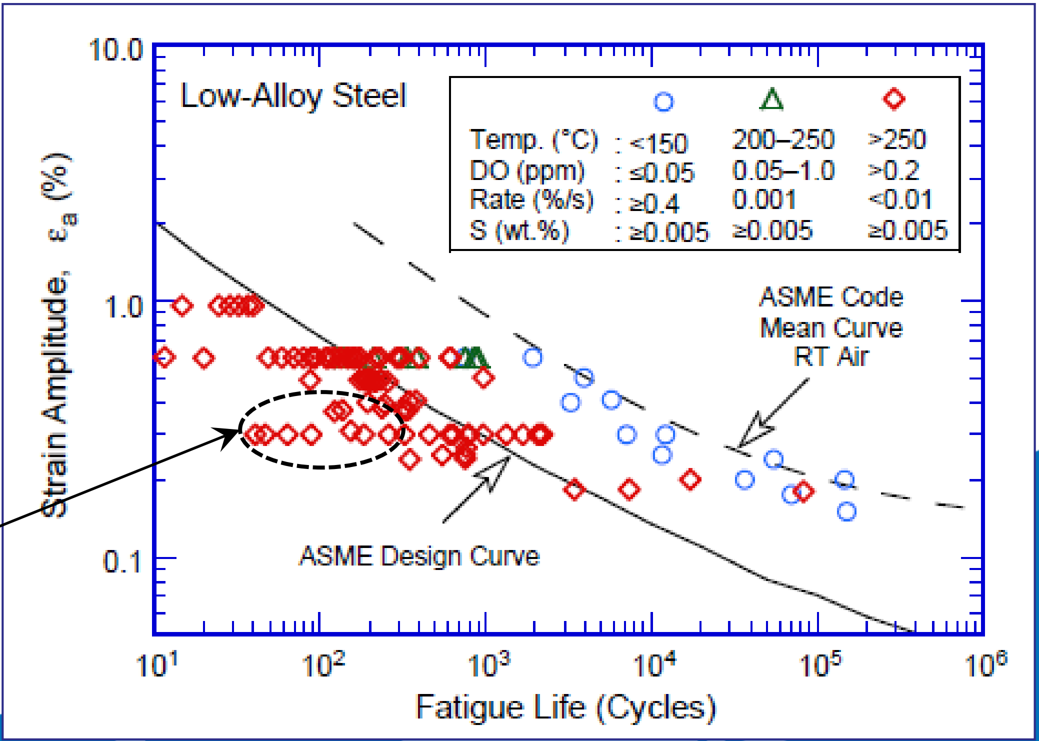
Example Fatigue Curve

Environmentally Assisted Fatigue

- ASME Code Section III fatigue curves developed from small-scale, polished specimens tested in **air**
  - Develop best-fit log-log curves for each material type
  - Adjust best-fit curves for worst-case mean-stress effects using modified Goodman relationship
  - Apply factors* of 2 on strain amplitude (e_a) or 20 on cycles (N), whichever is more conservative, to develop **air** design curves for each material

- Laboratory testing of specimens tested in water show that the air design curves may not adequately define fatigue life for materials exposed to water

*Some of the tests in **water** fall below the **air** design curve.*



Environmental Fatigue Correction Factor

- Environmental fatigue correction factor (**F_{en}**) is defined as the ratio of fatigue life in air at room temperature to the fatigue life in water:

$$F_{en} = N_{air}/N_{water}$$

- F_{en} is multiplicative to the calculated CUF in air:

$$CUF_{en} = U_1 F_{en,1} + U_2 F_{en,2} \dots U_Z F_{en,Z}$$

- From Revision 0 of NUREG/CR-6909 for stainless steel materials

$$F_{en} = \exp [0.734 - T' O' R']$$

where:

T' = transformed temperature:

$T' = 0$	for temperature, $T \leq 150^oC$
$T' = (T - 150)/175$	for $150 < T < 325^oC$
$T' = 1$	for $T \geq 325^oC$

O' = transformed oxygen:

$O' = 0.281$	for all fluid dissolved oxygen levels

R' = transformed strain rate:

$R' = 0$	for strain rate, $R \geq 0.4\%/s$
$R' = \ln(R/0.4)$	for $0.001 \leq R < 0.4\%/s$
$R' = \ln(0.001)$	for $R < 0.001\%/s$

Background:
Current NRC Guidance on EAF

Regulatory Guidance on EAF

- Initial NRC research efforts related to EAF
  - Chopra, O. K., and W. J. Shack, “Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low–Alloy Steels,” NUREG/CR–6583, 1998.
  - Chopra, O. K., “Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels,” NUREG/CR–5704, 1999.
- These NUREGs provided basis for guidance for license renewal applicants in the initial release of NUREG-1801, “Generic Aging Lessons Learned (GALL) Report” (2001)
  - Chapter X.M1, “Metal Fatigue of Reactor Coolant Pressure Boundary”
- Updated and consolidated EAF technical basis
  - Chopra, O. K., and W. J. Shack, “Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials – Final Report,” NUREG/CR–6909, Feb. 2007.

Regulatory Guidance on EAF, cont.

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- Operating reactors
  - Original licensing period: No guidance or requirements for considering EAF
  - License renewal period: Recent applicants use NUREG-1801, Rev. 2
    - Carbon steel: May use either NUREG/CR-6583, NUREG/CR-6909, or NRC-approved alternative
    - Stainless steel: May use either NUREG/CR-5704, NUREG/CR-6909, or NRC-approved alternative
    - Ni-Cr-Fe alloys: May use NUREG/CR-6909 or NRC-approved alternative
  - **Subsequent license renewal period: Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report (NUREG-2191)**
    - **All materials: May use NUREG/CR-6909, Rev. 0 (with correct average temperature), NUREG/CR-6909, Rev. 1 or NRC-approved alternative**
- New reactors
  - RG 1.207, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light Water Reactor Environment for New Reactors" March 2007.
  - Technical basis for RG 1.207 is NUREG/CR-6909
  - Both RG 1.207 and proposed Revision 1 of RG 1.207 use the F_{en} method summarized in Appendix A of NUREG/CR-6909.

Background:
Proposed Revision of NRC Guidance

Revision of Reg. Guide 1.207:
Draft for Public Comment

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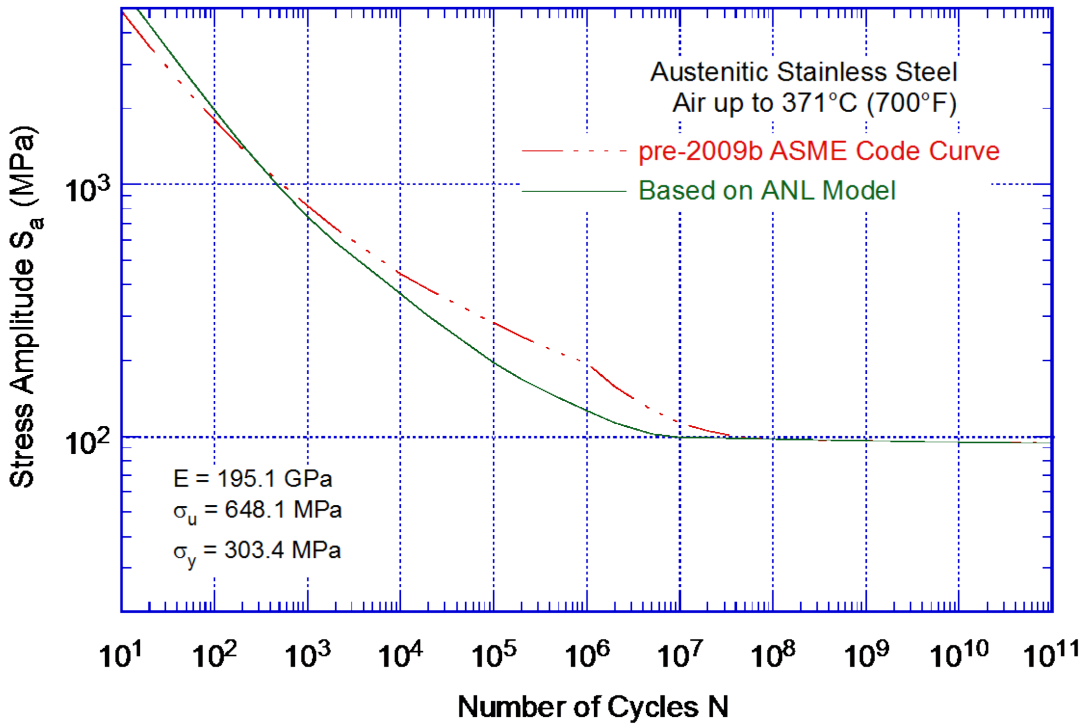
- Rationale for revision
  1. Consolidate all EAF guidance
  2. Update the guidance based on stakeholder feedback
  3. Update the guidance based on all available research data
- RG 1.207 significant changes
  1. The RG was made applicable to all LWRs
  2. The guidance was clarified to apply to all metal components exposed to LWR environments that have a CUF calculation required by a plant's current licensing basis (CLB)
  3. The background section was revised to incorporate the relevant content for operating reactors, license renewal, etc.
  4. The F_{en} equations were revised based on stakeholder feedback and the updated research as documented in NUREG/CR-6909, Rev. 1

Draft NUREG/CR-6909, Rev. 1:
Changes to Air Fatigue Curves

- Best-fit mean-data air curves ***are the same*** between NUREG/CR-6909, Rev. 0 and Rev. 1 for all materials
- Adjustment factors of 12 on cycles ***retained for consistency*** with Rev. 0 and ASME
  - **Requested public feedback on adjustment factors in FRN**
- Design curves ***are the same*** in Rev. 0 and Rev. 1 for carbon, low-alloy, and stainless steels
- Recommend use of stainless steel design curve for Ni-Cr-Fe alloys (conservative)



Draft NUREG/CR-6909, Rev. 1:
Significant Changes to F_{en}

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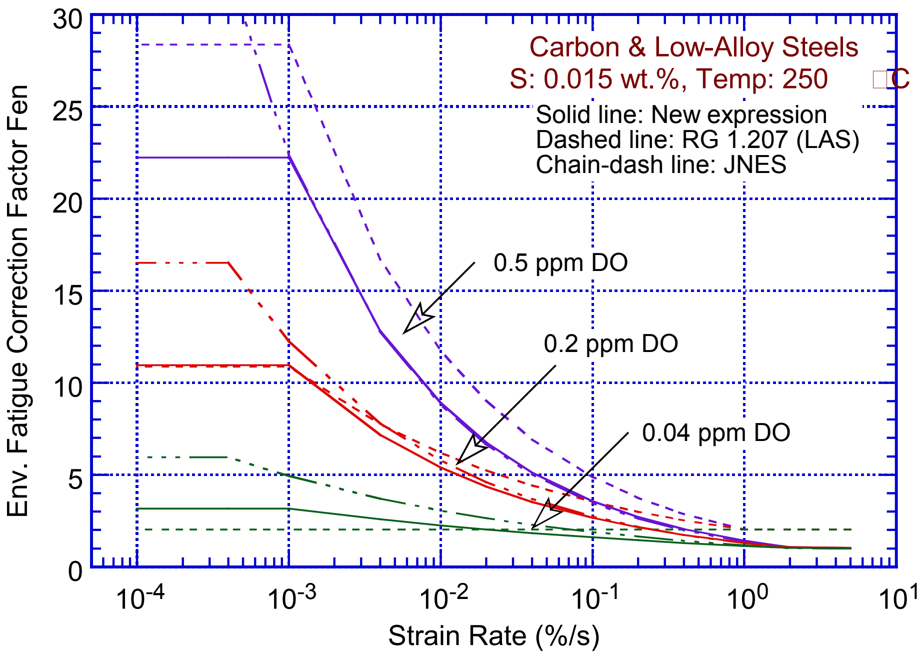
NUREG/CR-6909, Rev 0

- Different expressions for carbon and low-alloy steels
- F_{en} > 1.0 with no environmental effects
- Different constants in expressions for stainless steels and Ni-Cr-Fe alloys

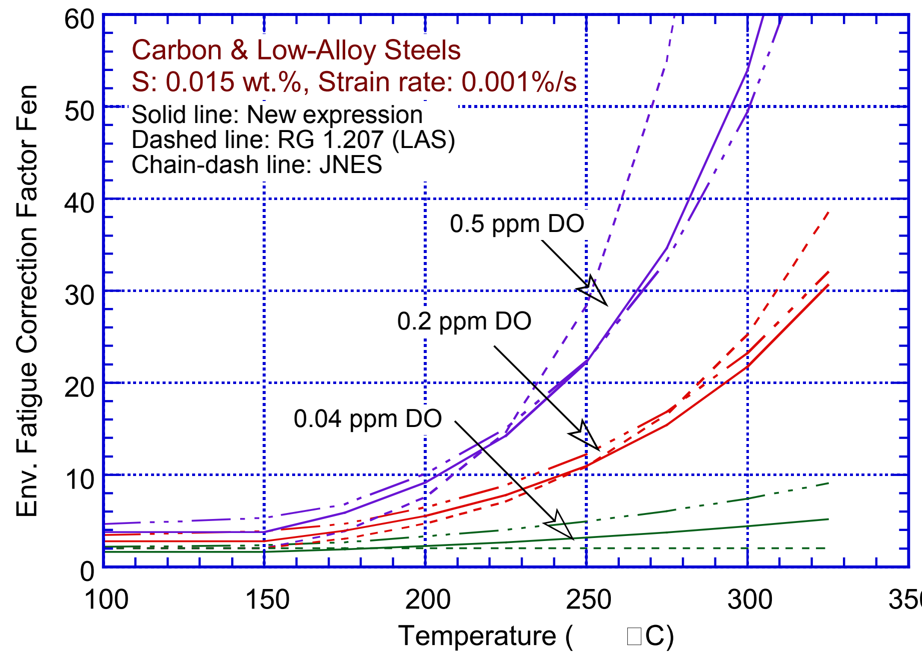
NUREG/CR-6909, Rev 1

- Same expression for carbon and low-alloy steels
- F_{en} = 1.0 with no environmental effects
- Same functional form for stainless steels and Ni-Cr-Fe alloys
- Expressions generally less conservative for all materials

F_{en} vs. Strain Rate, R



F_{en} vs. T



Draft NUREG/CR-6909, Rev. 1:
Other Significant Additions

- Validation calculations
  - Performed on both specimen and component tests (6 total test series)
  - Estimated life using ASME Code methods with F_{en} and compared prediction to experimental fatigue life
  - The predicted and measured lives for specimen tests agreed within the data scatter (i.e., factor of 2)
  - The predicted lives for component tests either agreed with or conservatively predicted the experimental results
- Sample problem (Appendix C)
  - Demonstrate one example application of the F_{en} methodology
  - Promote consistency in the application of EAF methods
  - Received positive feedback from stakeholders

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NUREG/CR-6909, Revision 1:
Overview of Public Comments

Overview of Public Comments

- Draft NUREG/CR-6909, Rev. 1 sent for comment under Federal Register Notice for docket ID NRC-2014-0023 dated April 17, 2014 (Vol. 79, No. 74) (FR Doc # 2014-08792)
  - Specifically asked for feedback on the following three areas:
    - I. The extension of the best-fit mean air curve for ferritic steels discussed in Section 3.1.10.
    - II. The air fatigue design curve adjustment factors summarized in Section 5.5.
    - III. Accuracy check of the technical content of the NUREG, particularly with respect to all of the numerical content of the report.
  - Public comment period ended on 6/2/2014.

Overview of Public Comments

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- Formal public comments were received from 10 commenters.

Letter No.	ADAMS Accession No.	Commenter Affiliation	Commenter Name	Abbreviation
1	ML14157A322	Consultant, Japan	Makoto Higuchi	HIGUCHI
2	ML14157A323	Consultant – CF Int. Engineering, France	Claude Faidy	FAIDY
3	ML14157A324	AMEC, United Kingdom	David Tice	AMEC
4	ML14157A325	Westinghouse Electric Company, USA	James Gresham	WEST
5	ML14157A326	Mitsubishi Heavy Industries, Japan	Seiji Asada	MHI
6	ML14157A327	Rolls Royce PLC, United Kingdom	Keith Wright	RR
7	ML14157A328	Electricite de France, France	Thomas Metais	EDF
8	ML14157A330	Hitachi, Japan	Akihiko Hirano	HITACHI
9	ML14157A331	AREVA, Inc., USA	Devin Kelley	AREVA
10	ML14157A332	Kansai Electric Power Company, Republic of Korea	June-soo Park	KEPCO

- Three additional commenters provided feedback after the public comment period officially ended.

Overview of Public Comments

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- Comments enumerated (if they weren't already) and then partitioned into single issues (sub-comments) wherever possible
- Each issue was uniquely identified and tracked as [XXX]-[YYY]-[ZZZ][a]
  - [XXX] = Abbreviation from table
  - [YYY] = Letter No. from table
  - [ZZZ] = Sequential comment number
  - [a] = sub-comment
- Total sub-comments/Issues: 254
  - Formal commenters 235
  - Additional commenters 5
  - Authors and staff comments 14

Overview of Public Comments

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- Comments were generally technical in nature, thoughtful, and often expansive
- Most technical comments associated with the following areas
  - Scope of the F_{en} method
  - Adjustment factor analysis and application
  - Clarification of statements
  - Relevance to nuclear plant applications
  - Overall conservatism of ASME requirements in conjunction with F_{en}
- Staff and authors agree with over 95% of the individual sub-comments
- Areas of disagreement are generally not significant with respect to F_{en} method.
  - Interpretations of ASME Code requirements
  - Application of load sequence effects
  - Basis for and application of strain threshold
  - Interpretation of AREVA test results
  - High-cycle cut-off of design curve

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NUREG/CR-6909, Revision 1:
Sample Public Comments and Responses

Sample Comments and Responses:
Adjustment Factors

- Comment
  - *All the reduction factors are considered independent, it's not accepted in all international approaches in particular a constant F_{en} independently of number of cycles is not justified [stet] clearly* (FAIDY-2-3a)
- Summary of response
  - The report does consider the adjustment factors to be independent
    - There is insufficient data to develop correlation factors for a more rigorous analysis.
    - This point has been clarified in the NUREG..
  - The method presented does assume that F_{en} is not a function of applied strain.
    - The point has been clarified several places within the NUREG.

Sample Comments and Responses:
High Cycle Fatigue

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- Comment
  - *"…Extension of the Best-Fit Mean Curve from 10^6 to 10^{11} cycles …is too conservative."* (MHI-5-1a)
- Summary of response
  - Extension of the fatigue curve is conservative.
  - NUREG/CR-6909 Rev. 1, extension is identical to that proposed by the ASME Code committees.
  - Extension is based on data that has a prominent mean stress component.
  - Basing the curve on such data is meant to be both conservative and also allow application to engineering components, which often have high mean stress loading.
  - NUREG states that, for data without a significant mean stress effect, this proposed curve could be significantly conservative.

NUREG/CR-6909, Revision 1:
Changes to Document

NUREG/CR-6909, Revision 1:
Modifications After Public Comment

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- Authors and staff made significant modifications to the NUREG in an attempt to address virtually all of the public comments
  - Explain more clearly and completely all the technical bases and assumptions supporting the work.
  - Summarize the current state of knowledge
  - Provide a foundation for continued research
- NUREG has expanded significantly over time
  - Rev 0: 120 pages
  - Rev 1, draft: 320 pages
  - Rev 1, final: almost 500 pages
    - Revised Main Body – +10 pages
    - New Appendix D – 135 pages
    - New Appendix E – 12 pages

NUREG/CR-6909, Revision 1:
Major Changes due to Comments

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- Added new Section 1.5 on bases and assumptions of F_{en} method
- Moved original Section 4.1.14 (Modified Rate Approach) to new Section 4.4 and clarified write-up.
- Reworked and revised example problem (Appendix C)
- Eliminated all Revision 0 equations from main body
- Added App. E: “Equations in NUREG/CR-6909 Rev. 0 and Rev. 1”
  - Equations and Equation Numbers from NUREG/CR-6909 Rev. 0.
  - Equations and Equation Numbers in NUREG/CR-6909 Rev. 1
  - Changes in the Equations or their Number in NUREG/CR-6909 Rev. 0 and Rev. 1
- Replaced all figures in main body with higher resolution images
- Added App. D: “Compendium of Figures”
  - Enlarged and high resolution images of all figures in main body
- Defined LWR water environment and changed “reactor coolant” to “water” throughout report, as appropriate
- Subjected NUREG to technical editing

Revision 1 of Reg. Guide 1.207:

Overview of Public Comments

Overview of Public Comments

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- Revision 1 of RG 1.207 (DG-1309) sent for comment under Federal Register Notice for docket ID NRC-2014-0244 dated November 24, 2014 (Vol. 79, No. 226) (FR Doc # 2014-27712)
- Public comment period ended on 1/23/2015.

Overview of Public Comments

U.S.NRC
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- Formal public comments were received from 7 commenters.

Letter No.	ADAMS Accession No.	Commenter Affiliation	Commenter Name	Abbreviation
1	ML15023A569	Rolls Royce PLC, United Kingdom	Keith Wright	RR
2	ML15023A570	Westinghouse Electric Company, USA	Camille Zozula	WEST
3	ML15023A571	Nuclear Energy Institute, USA	Jason Remer	NEI
4	ML15027A334	Union of Concerned Scientists, USA	David Lochbaum	UCS
5	ML15033A382	Southern Nuclear Operating Company, Inc., USA	Charles Pierce	SNOC
6	ML15033A383	AREVA, USA	Morris Byram	AREVA
7	ML15033A384	Electric Power Research Institute, USA	Nathan Palm	EPRI

- Four organizations (Rolls Royce, Westinghouse, AREVA, and EPRI) commented on both NUREG/CR-6909, Revision 1 and DG-1309
- Each comment was tracked as [XXX]-[YYY]-[ZZZ]
  - [XXX] = Abbreviation from table
  - [YYY] = Letter No. from table
  - [ZZZ] = Sequential comment number
- Total Comments/Issues: 49

Overview of Public Comments

U.S.NRC
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- Almost all comments associated with the following areas
  - Editorial or clarification (≈ 45%)
  - NUREG/CR-6909, Rev. 1 technical basis (≈ 22%)
  - Applicability of earlier technical reports and guidance (≈ 14%)
  - RG scope, use, and applicability (≈ 12%)
  - Miscellaneous (≈ 6%)
- Staff fully agree with about 1/2 of the comments
- Staff partially agree with about 1/4 of the comments
- Most common areas of disagreement
  - Applicability of earlier technical reports and guidance
  - RG scope, use, and applicability

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Revision 1 of Reg. Guide 1.207:
Sample Public Comments and Responses

Sample Comments and Responses:
Applicability of Prior Guidance

- Comment
  - *"…the DG does not clarify if the use of NUREG/CR-6909, Revision 0 formulas remains acceptable.*
  - *Several LR applicants have used NUREG/CR-6909, Revision 0 methods and formulas for computing F_{en} values and would not wish to revise them just in order to meet NUREG/CR-6909, Revision 1 criteria. (NEI-3-1)*
- Summary of response
  - Staff disagrees that guidance for prior methods should be included within the RG
  - Prior methods previously approved by the staff remain valid for the period of their intended use
  - Staff is finalizing SLR-specific guidance which clarifies the use of prior methods for SLR.

Sample Comments and Responses:
Scope, Use, and Applicability of RG

- Comment
  - *Page 6, paragraph 2, states: "These methods apply to those components exposed to reactor coolant that are required by regulation to have a fatigue CUF evaluation or have an existing CLB fatigue CUF evaluation."*
  - *There are components that 'have an existing CLB fatigue CUF evaluation' in secondary systems. They are not required by regulation to have a fatigue CUF. The applicability of F_{en} to such components should be clearly stated (WEST-2-3)*
- Summary of response
  - Applicable environments
    - RG is applicable to both primary and secondary systems
    - Draft RG does not clearly define the terms "reactor coolant" and "coolant"
    - Replaced "coolant" with "water"; added definition for LWR water environment
  - Clarified RG applicability for licensing actions associated with
    - Reactor designs submitted for NRC approval
    - Operating reactors pursuing license renewal
    - Plants where addressing such the effects of the LWR water environment is part of their CLB

Revision 1 of Reg. Guide 1.207:

Changes to Document

Summary of RG 1.207 Revisions:
Major Changes Resulting from Comments

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- Defined LWR water environment and changed "reactor coolant" and "coolant" to "water" throughout RG, including title
- Clarified applicability of RG
    - Reactor designs submitted for NRC approval
    - Operating reactors pursuing license renewal
    - Plants where addressing such the effects of the LWR water environment is part of their CLB
- Clarified that guidance is not applicable for Inconel 718
- Clarified that Ni-Cr-Fe alloys should use the stainless steel design curves in air provided in NUREG/CR-6909, Rev. 1 (or associated ASME Code Section III curves)

Current Status and Next Steps

Guidance Finalization:
Current Status and Next Steps

- Revision 1 of RG 1.207
  - Completed technical concurrence (December 2016)
  - Some changes to RG and responses to public comments since ACRS Metallurgy and Fuels Subcommittee Briefing (12/15/16)
    - No technical changes
    - Several editorial changes
    - Applicability statements in Parts B. and C.
- NUREG/CR-6909, Revision 1
  - Incorporated technical editing changes
  - Conforming changes to responses to public comments are needed
  - A few minor changes resulting from concurrence process
- Requesting ACRS recommendation to finalize RG 1.207
- RG 1.207 should be finalized before issuance of SLR guidance (i.e., NUREG-2191, NUREG-2192) in mid-2017.
  - Planning to finalize RG in March 2017

Back-up Slides

Related Regulatory Requirements

Title 10 of the Code of Federal Regulations (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities, Appendix A, “General Design Criteria for Nuclear Power Plants”

- General Design Criterion 1

  Safety related SSCs must be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function performed

- General Design Criterion 30

  Components included in the reactor pressure boundary must be designed, fabricated, erected, and tested to the highest practical quality standards

- 10 CFR 50.55a (c), endorses ASME Code for design of safety-related systems and components (Class 1)
  - ASME Code, Section III fatigue design curves
  - Fatigue design curves do not address the impact of the water environment
- **Regulatory Guidance on EAF exists to provide an acceptable method for addressing the impact of water environment on fatigue calculations**

Updated Air Fatigue Data

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Material	Data in NUREG/CR-6909, Rev. 0	Data in NUREG/CR-6909, Rev. 1	Increase*
Carbon Steels	153 points (8 heats) [Figure 7(a) of Rev. 0]	254 points (19 heats) [Figure 32(b) of Rev. 1]	66 %
Low-Alloy Steels	358 points (19 heats) [Figure 7(b) of Rev. 0]	430 points (22 heats) [Figure 32(d) of Rev. 1]	20 %
Austenitic Stainless Steels	357 points (38 heats) [Figure 35 of Rev. 0]	622 points (40 heats) [Figure 45(b) of Rev. 1]	74 %
Ni-Cr-Fe Alloys	Not quantified [Figures 56 & 57 of Rev. 0]	559 points (45 heats) [Figures 50 – 52 of Rev. 1]	N/A

* The majority of additional data from Report No. JNES-SS-1005.

* NRC gratefully acknowledges the release of the Japanese EAF research data. The NUREG revisions would not have been as comprehensive without this information.

Updated Water Fatigue Data

Material	Data in NUREG/CR-6909, Rev. 0	Data in NUREG/CR-6909, Rev. 1	Increase*
Carbon Steels	318 points (12 heats) [Figure 27 of Rev. 0]	638 points (21 heats) [Figure 79 of Rev. 1]	100 %
Low-Alloy Steels	327 points (13 heats) [Figure 27 of Rev. 0]	536 points (20 heats) [Figure 79 of Rev. 1]	64 %
Austenitic Stainless Steels	276 points (14 heats) [Figure 52 of Rev. 0]	683 points (32 heats) [Figure 108 of Rev. 1]	147 %
Ni-Cr-Fe Alloys	Not quantified [Figures 58 & 59 of Rev. 0]	162 points (13 heats) [Figures 109 – 110 of Rev. 1]	N/A

* The majority of additional data from Report No. JNES-SS-1005.

* NRC gratefully acknowledges the release of the Japanese EAF research data. The NUREG revisions would not have been as comprehensive without this information.

Sample Comments and Responses:
ASME Method and Requirements

- Comment
  - *Hasn't been a "…reconciliation between the specimen fatigue test data… and the complete ASME-Code Fatigue Methodology ….."*
  - *"…ASME-Code Fatigue Methodology already contains a lot of multiplication of effects that have not been considered by the developers of both the in-air design fatigue curves and the F_{en} factors."* (AREVA-9-17f)
- Summary of response
  - New Section 1.5 identifies and discusses some ASME Code conservatisms.
  - The F_{en} approach, in concert with ASME design curve, will lead to either accurate or conservative predictions of environmental effects.
  - Eliminating conservatism in the ASME Code is outside the report scope.
  - A technical basis for revising ASME Code procedures could be developed by working through the appropriate Code committees

Sample Comments and Responses:
Miscellaneous

- Comment
  - *"…the only concerns we should have are for those transients with low strain rates (slow transients)."*
  - *The other transients (for example, in-surges and out-surges) need to be evaluated for fatigue, but the current ASME-Code Class 1 Component classic Fatigue Methodology (without any F_{en} penalty factors) is very appropriate for those transients…"* (AREVA-9-17k)
- Summary of response
  - Comment postulates that environmental effects should only be considered for slow transients.
  - Most transients have a wide range of strain rates and importance of environmental effects is not always obvious without evaluating their effects in totality.
  - Entire transient has to be evaluated to determine both the F_{en} and the accumulated strain associated with each strain rate range.
  - The average F_{en} for the transient can then be determined.

NUREG/CR-6909, Revision 1:
Major Sections - After Public Comments

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Summary of RG 1.207:
Outline of Unique Sections

A. Introduction
   - Purpose
   - Applicable Rules and Regulations
   - Related Guidance

B. Discussion
   - Reason for Revision
   - Background

C. Regulatory Position

   1. Carbon and Low-Alloy Steels and Welds
      1.1 CUF in Air
      1.2 Environmental Factor (F_{en})
      1.3 Environmental CUF

   2. Wrought and Cast Austenitic Stainless Steels and Welds

   3. Ni-Cr-Fe Alloys and Welds

D. Implementation

- References

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Quality Assurance Program Implementation under 10 CFR Part 52

ACRS Full Committee

February 9, 2017

Jermaine Heath
Quality Assurance Vendor Inspection Branch
Division of Construction Inspection & Operational Programs
Office of New Reactors

Presentation Outline

- Background
- Quality Assurance (QA) Program Implementation for New Reactors
  - DC \ COL Applicant Responsibilities
  - NRC QA Licensing Review
  - NRC QA Inspection Programs
- Conclusions
- Discussion/Committee Questions

DC - Design Certification, COL – Combined License

Background

- During its April 2016 meeting (633rd), the Advisory Committee on Reactor Safeguards (ACRS) reviewed five exemption requests for the AP1000 certified design that Duke included in its Levy County Combined License Application (COLA). The departures would be made common to all COLAs referencing the AP1000 design

- ACRS recommended that staff evaluate any lessons learned, relative to ongoing and future oversight of the quality assurance program implementation during development of designs seeking certification under 10 CFR Part 52

Background

- NUREG-1055, “Improving Quality and the Assurance of Quality in the Design and Construction of Nuclear Power Plants”
  - QA problems were the result of utilities’ ineffective implementation of QA
  - NRC's past licensing and inspection practices did not adequately screen construction permit applicants
- QA lessons learned from NUREG-1055 were incorporated into Part 52 licensing process
- NRC current processes involve more QA inspections during DC process

QA Program Implementation for New Reactors

<u>DC / COL Applicant Responsibilities</u>

- Appendix B to 10 CFR 50 **applies** to the development of safety-related information reflected in a certified design under 10 CFR Part 52
- Must describe how Appendix B requirements are met
- For DC applicants (Part 52 Subpart B)
  - 10 CFR 52.47(a)(19)
  - Quality Assurance Program Description (QAPD) should address design QA activities in support of a DC, not construction and design QA activities once construction begins
- For COL applicants (Part 52 Subpart C)
  - 10 CFR 52.79(a)(25)
  - QAPD should address all phases of a facility's life, including design, construction, and operation

QA Program Implementation for New Reactors

DC / COL Applicant Responsibilities

- **Appendix B to 10 CFR Part 50**

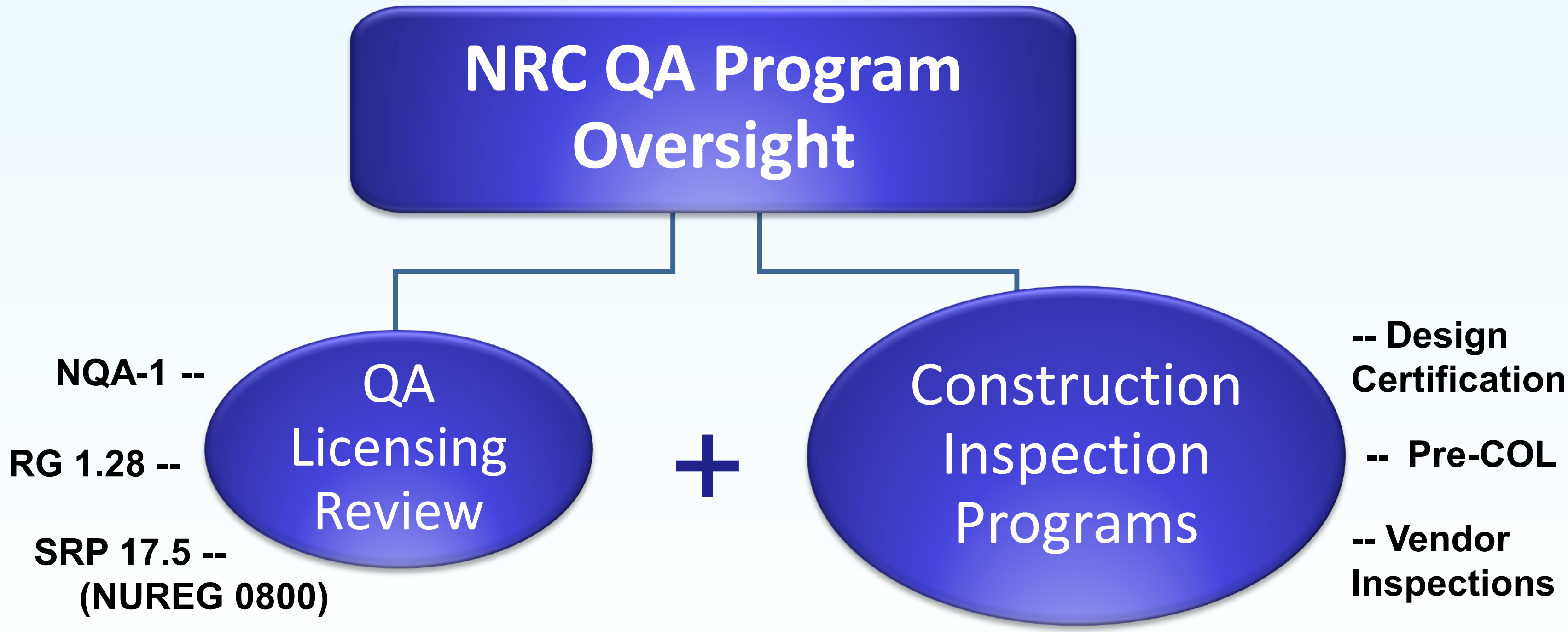
Criterion I, Organization
- Retain responsibility for the QA program

Criterion III, "Design Control"
- Establish organizational responsibilities
- Detail design inputs & analysis
- Translate design requirements into procedures
- Establish design interface controls (internal\external)
- Provide suitable record keeping.

Criterion VII, "Control of Purchased Material, Equipment, Services"
- Verify conformance of purchased safety-related items and services
- Assess control of quality by contractors at intervals

Criterion XVIII "Audits"
- Conduct periodic audits to verify compliance with App. B. (internal/external).

QA Program Implementation for New Reactors



Legal Authority for Conducting Inspections under Part 52

How is compliance with Appendix B verified prior to a DC applicants submittal of a Part 52 application?

- Applicant retains responsibility for implementation of QA program
- No NRC regulatory basis to conduct pre-application QA inspections prior to docketing
- NRC construction inspection program is implemented when:
  - (1) QAPD is docketed; **AND**
  - (2) 10 CFR Part 21 invoked through purchase order for safety-related services or components

QA Program Implementation for New Reactors

<u>NRC Construction Inspection Program</u>

- Design Certification (IMC 2508)
  - Applies when applicant submits DC application
  - QA program review
  - Post-Docketing QA Program Inspection (IP 35017)
  - Design Qualification Testing Inspection (IP 35034)
- Pre-COL Phase (IMC 2502)
  - Applies when applicant submits COL application
  - Implemented prior to license issuance
  - Post-Docketing QA Program Inspection (IP 35017)
  - Oversight of Pre-construction activities (IP 35007)

IP = Inspection Procedure
IMC = Inspection Manual Chapter

QA Program Implementation for New Reactors

NRC Construction Inspection Program

- Review of Detailed Design Development
- Vendor Inspection Program (IMC 2507)
  - IP43002, Routine Inspections of Nuclear Vendors
  - IP43003, Reactive Inspections of Vendors
  - IP43004, Inspection of Commercial-Grade Dedication Programs
  - IP36100, Inspection of 10 CFR Part 21 and Programs for Reporting Defects and Noncompliance
  - IP 37805, Engineering Design Verification Inspection

The terms “vendor,” and “supplier” are used interchangeably

QA Program Implementation for New Reactors

NRC Inspection of the Design Authority

- IP 37805 – Eng. Design Verification Inspection (1600 hrs)
  - Conducted when detailed design is ~70% complete
  - Risk-informed sample, detailed review of selected systems
  - Verifies design authority has developed processes that allow for the complete and accurate transfer of the high level design and performance requirements specified in FSAR into detailed procedures, and specifications
  - Verifies design changes are adequately controlled

QA Program Implementation for New Reactors

Conclusions

- Quality assurance is integral to nuclear power plant design and construction
- Lessons learned from NUREG 1055 are still relevant today as they relate to QA design and construction
- DC / COL applicant retains responsibility for the establishment and execution of the QA program, while NRC provides oversight of its implementation
- NRC acceptance of an applicant's QA program ensures that adequate controls are in place to meet the regulatory requirements of Appendix B
- The current QA licensing review process and inspection programs are effective and we continue to review and update staff guidance on licensing reviews and inspection

QA Program Implementation for New Reactors

Questions & Discussion

QA Program Implementation for New Reactors

Supplemental Slides

BACKGROUND: ACRS Concerns Regarding Appendix B and DC Pre-applicants

Background

- During its April 2016 meeting (633rd), ACRS reviewed 5 exemption requests for WEC AP1000 certified design that Duke included in its Levy County COLA. The departures would be made common to all COLAs referencing the AP1000 design, and current COL holders.

NRC Staff Actions

- In a May 2016 letter to the ACRS (ML16117A447), the EDO acknowledged the NRC's statutory limitations in dealing with organizations that have not formally submitted [DC] applications for NRC review. The staff intends to address the following ACRS interests:
    1. Does 10 CFR 50, Appendix B, apply to the development of safety-related information reflected in the certified design? (Slide 5)
    2. If yes, and given the statutory limitations which exist prior to application submittal, how is compliance with Appendix B during this period verified? (Slide 8)
    3. If no, how are Appendix. B requirements met for safety-related information reflected in the certified design? (N/A)
    4. Whether yes or no, who is responsible for verification that Appendix B requirements have been met for safety-related information reflected in the certified design? (Slide 6)
    5. Does a COL applicant have any responsibility for verification that Appendix B requirements are met: (a) for safety-related information reflected in the certified design (Slide 7) or (b) for safety-related information developed by the certified design holder to implement the certified design? (Slide 6)
    6. If the NRC is responsible for verification that Appendix B requirements are met for safety-related information reflected in the certified design, but a COL applicant/holder is responsible for verification that the Appendix B requirements are met by the certified design holder for safety-related information to implement the certified design, how is this transition in responsibility identified and implemented? (Slide 7)

Regulations and Guidance

Quality Assurance Requirements

- 10 CFR 50.34 (a)(7) requires submittal of a description of the quality assurance program to be applied to the design, fabrication, construction, and testing of the structures, systems, and components (SSCs) of the facility and a discussion of how the applicable requirements of Appendix B will be satisfied
- 10 CFR 52.47(a)(19) and 10 CFR 52.79(a)(25) require a DC or COL applicant to include a QAPD to be applied to the design, fabrication, construction, and testing of the SSCs of the facility.
- RG 1.28, “Quality Assurance Program Criteria (Design and Construction),”
- Appendix B to 10 CFR Part 50, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants”
- Regulatory Guide 1.206, Section C.III.1, Chapter 17, Section C.III.17.5, “Quality Assurance Program Guidance,” Revision 0, June 2007
- NUREG-0800, Standard Review Plan (SRP), Section 17.5, “Quality Assurance Program Description – Design Certification, Early Site Permits, and New License Applicants,” Revision 0, March 2007
- Inspection Manual Chapter 2506, “Construction Reactor Oversight Process General Guidance and Basis Document,” dated March 16, 2015

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