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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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659TH MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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THURSDAY

DECEMBER 6, 2018

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ROCKVILLE, MARYLAND

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The Advisory Committee met at the Nuclear Regulatory Commission, Three White Flint North, Room 1C3 & 1C5, 11601 Landsdown Street, at 1:00 p.m., Michael L. Corradini, Chairman, presiding.

COMMITTEE MEMBERS:

MICHAEL L. CORRADINI, Chairman

PETER RICCARDELLA, Vice Chairman

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DENNIS C. BLEY, Member

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JOSE MARCH-LEUBA, Member
HAROLD B. RAY, Member
JOY L. REMPE, Member
GORDON R. SKILLMAN, Member
MATTHEW W. SUNSERI, Member

ACRS CONSULTANT:
STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIALS:
QUYNH NGUYEN
KENT HOWARD

ALSO PRESENT:
KENNETH BROWNE, NextEra
WILLIAM BURTON, NRR
ANDY CAMPBELL, NRO
EDWARD CARLEY, NextEra
MICHAEL COLLINS, NextEra
JOSEPH DONOGHUE, NRR
ALLEN FETTER, NRO
RUDY GIL, NextEra
MICHELLE HART, NRO
ALLEN HISER, NRR

ARCHIE MANOHARAN, Tennessee Valley Authority
ERIC McCARTNEY, NextEra
BRUCE MUSICO, NSIR
ERIC OESTERLE, NRR
RAYMOND SCHIELE, Tennessee Valley Authority
MICHAEL SCOTT, NSIR
DANIEL STOUT, Tennessee Valley Authority
MALLECIA SUTTON, NRO
ALEX YOUNG, Tennessee Valley Authority

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

P R O C E E D I N G S

(1:00 p.m.)

CHAIRMAN CORRADINI: Okay. The meeting will come to order. This is the first day of the 659th meeting of the Advisory Committee on Reactor Safeguards.

During today's meeting the Committee will consider the following. Clinch River early site permit, Seabrook License Renewal Application, and then preparation of ACRS reports.

The ACRS was established by statute, and is governed by the Federal Advisory Committee Act, or FACA. As such, this meeting is being conducted in accordance with the provisions of FACA. That means that the Committee can only speak through its published letter reports.

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

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

Mr. Quynh Nguyen is the designated Federal Official for the initial portion of the Meeting. The ACRS section of the US NRC public website provides our charter, by-laws, letter reports, and full transcripts of all our full and Subcommittee meetings, including all the slides presented at those meetings.

At this time we've not received any written comments, or requests to make oral statements from members of the public regarding today's session. There will be phone bridge line. To preclude interruption of the meeting the phone will placed in a listen in only mode during the presentation of the Committee discussion.

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

So, at this time I'll just remind everybody, take all your things and turn them off, or put them in mute, so we don't have to hear buzzing or beeping. And with that I'll turn to Member Kirchner to lead us through the first topic.

MEMBER KIRCHNER: Thank you, Chairman. Apologies for the slight delay in arriving. We have heard from the applicant and staff over the course of the last year.

We had a informational briefing on November 15th of last year. Then we had four additional informative meetings with both parties. So, with that I'm ready to turn it over to the staff, to Andy Campbell to proceed, please.

MR. CAMPBELL: If I can remember how to turn these things on. I'm Andy Campbell. I'm the Deputy Director of the Division of Licensing, Siting, and Environmental Analysis in the Office of New Reactors at the NRC.

Mr. Chairman, it is a great pleasure to be here today for the full Committee meeting on the Clinch River Nuclear site, early site permit, what we'll call the SP, application safety review submitted to the NRC May 26, 2016.

This submittal is the first ESP for a small modular reactor plant design. And it was prior to staff's work on the small modular reactor and other new technologies rulemaking. Accordingly, the application and the review of the application by the staff is based on current regulations and guidance.

Staff has presented a series of ACRS Subcommittee meetings on the staff's safety review of the application. And today staff will be presenting our final overview, with no open items, for the Clinch River ESP safety evaluation report.

The ESP review has been progressing consistent with the schedule, and completion of today's full Committee now puts the project ahead of schedule.

For example, staff provided an overview to ACRS in November 15, 2017, a little over a year ago. Previous staff presentations for the relevant SER chapters to several ACR Subcommittee meetings, from May 15 of this year, 2018, to November 14, 2018.

The NRC staff safety review of the application included the execution and completion of five audits and one inspection, and the issuances of 12 RAIs comprising 50 questions.

The staff completed all the advance safety evaluation with no open items.

Staff's presentation, and then the applicant's presentations today are, we're going to focus on, the staff will focus on the EPZ, with an overview of the other Subcommittee presentations.

One key point is, if the exemptions are approved for the ESP, the COL applicant can adopt these exemptions if it shows that a COLA PEPE EPZ source term release to the atmosphere are bounded by the non-design specific plant parameter source term information developed for the ESP.

A future COL application featuring an SMR design that fits within the plant parameter envelope established in the ESP could apply the approved methodology to the design selected, to determine the appropriate PEP EPZ, and for the site, and also to demonstrate whether the conditions for either of the two sets of exemptions have been met.

Also in the audience today, besides NRC staff and applicant staff are representatives from the Federal Emergency Management Agency, FEMA, Technological Hazards Division. And representatives from Tennessee Emergency Management Agency are on the conference bridge. So, now I'm going to turn it back to you.

MEMBER KIRCHNER: So, thank you. I think we're going to turn to the applicant at this point. Okay. Dan, please proceed.

MR. STOUT: Thank you. Good afternoon. I want to start by expressing our appreciation for the flexibility to adjust the schedule, and get this done. We took advantage of the opportunity and got to go pay our respects at the Capital yesterday morning early.

And so, win, win.

So, I'm Dan Stout. I'm the Director of Nuclear Technology and Innovation for the Tennessee Valley Authority, managing this small module reactor activity, particularly the early site permanent application.

I'll be kicking off an introduction, talking about the site and the SMR program. And then I'm going to turn it over to Ray Schiele, Licensing Manager, who's going to cover the specifics of the early site permanent application itself. And then, as requested, Archie Manoharan will be doing a deeper dive into the emergency preparedness portion of the application.

So, I'd like to acknowledge the Department of Energy, who has been an integral partner in supporting the SMR activities that TVA is undertaking, particularly with financial assistance. However, the views expressed are TVA's alone.

So, on Slide 5, I'll remind everyone that Tennessee Valley Authority's mission is broader than just making electricity. It's also important to be a good steward of the environmental resources, and to be a partner in economic development.

TVA has been focused on the Clinch River site in Oak Ridge, Tennessee. It's a 1,200 acre site. The project is confined to 335 acres on that 1,200 acre reservation.

And it is a good site. Has access to both 500 and 161 KV transmission, which cut through the site. It is a neighbor to the Department of Energy, a customer that is interested in the output from this project.

The site was disturbed back in the 1970s and '80s. It was the site of the former Clinch River breeder reactors. So, there's some basic infrastructure, roads, storm water retention, things like that.

The community of Oak Ridge, you couldn't ask for a better place to want to do something nuclear. Not only is there strong community support, but there's an abundant and skilled nuclear workforce there. And it's a site that's within TVA's ownership and control. So, it makes proceeding rather easy.

Next. So, the early site permit application itself consists of site safety analysis report, environmental report, Part 5 emergency plans. And we actually submitted two different emergency plans, one for site boundary, one for two mile. Archie will get into those details. And a consistent set of exemptions that go along with those emergency plans.

Our early site permit application is based upon a plant parameter envelope that was informed by the designs of the four U.S. light water reactors that were under development over the previous few years. That includes the B&W mPower, Holtec, NuScale, and Westinghouse.

The application was developed, and the plant parameter envelope was developed based upon NE 1001 guidance. It assume that two or more reactors of the same design deployed, and a maximum of 800 megawatts thermal for an individual reactor, and a maximum of 2,420 megawatts thermal for the site.

Next. So, the schedule, we're here focused on the safety element, which is the, kind of the top row. There's the other track, environmental, and then the hearings.

So, on the safety side the NRC schedule calls for issuance of the final safety evaluation report in August. We're hopeful that we're ahead of that schedule.

The environmental, the staff issued the draft environmental impact statement in April. And it looks like we're on track to be ahead of the June schedule goal.

On the hearing side there were four contentions filed. Two were admitted. In July the Atomic Safety and Licensing Board dismissed all outstanding contentions, and terminated the contested hearing. Subsequently, the Commission indicated that it's their intent to run the mandatory hearing.

Next. So, I'd like to hit some highlights of the early site permit application, and the review process itself. The NRC commenced the review in the very beginning of 2017. The application as originally submitted had about 8,000 pages, supported by about 80,000 pages of technical information.

One of the highlights I'd like to point out is the efficient use of audits. The staff did a great job of preparing well in advance, and listing out all of their questions, all of the information needs, well in advance of the audit.

So then, when the audit occurred we were able to prepare responses to all of those open items well, all of those information needs well in advance, so that when they were there face to face there was meaningful discussion on the challenges.

By the end of the audits we had clarity on how to resolve all the issues. That manifested itself in very few RAIs. As Andy mentioned, it's about a dozen, as compared to hundreds for prior applications.

And I'm going to attribute a lot of that success to very frequent, clear, and candid communication. We, both staff and the applicant, we identified issues early, and we escalated them, put the resources on those issues early.

So, next. So, I'd like to turn it over to Ray Schiele now to talk about the early site permit application.

MR. SCHIELE: Thank you, Dan. Good afternoon. I'm Ray Schiele, currently the Licensing Manager for the Clinch River Nuclear Early Site Permit Application. I have 44 years in this industry, primarily operations and licensing. And since 2016 the Licensing Manager for the Clinch River project.

Quick overview of the organization of the application. The Clinch River application contains the information required by 10 C.F.R. 52.17, contents of applications for an early site permit. And was submitted in accordance with NRC guidance on electronic submittals.

Part 1, administrative information. This section contains an overview of the early site permit application, a general description of the format, content of the application, and corporate information, including ownership, management, and Board of Directors.

Part 2, the SSAR, includes a discussion of the site description, safety assessment, quality assurance, general location of the site, site suitability, design parameters postulated for the CRN site, population profiles, and an assessment of site features that may affect the design chosen for the facility.

Part 3, environmental report. The ER addresses the environmental impacts associated with construction and operation of new SMRs.

Part 4, site redress plan. TVA is not pursuing a limited work authorization with this application. Therefore, there is no redress plan.

Part 5, emergency planning information. The emergency planning information includes major features of the emergency plan. And there will be more information with Archie.

Part 6, exemptions and departures. This part lists applicant requested exemptions that are authorized by law, would not endanger life, property, or common defense and security, and are otherwise in the public interest. A discussion and justification for each of the requests is included in this part. There were no departures requested in Part 6.

Part 7, withheld information. This part contains information redacted from other parts of the application due to sensitive or proprietary nature of the information.

And last, Part 8, enclosures. All enclosures submitted with the early site permit application are provided in Part 8.

ESPA development, the regulatory bases. This slide illustrates the regulatory bases for the development of both the SSAR and ER. The regulatory bases consist of various regulations, standard review plans, reg guides, and review standards.

NRC interactions. Prior to the ESPA submittal in May of 2016 the NRC performed pre-application site visits, alternative site visits, and pre-application readiness review.

After submittal the NRC performed three major audits in the spring and summer of 2017, supporting hydrology, ground water, seismic, geotech, and environmental.

In addition, a comprehensive four month EP audit not listed on this slide commenced in the fall of 2017, and was supplemented by an additional audit in the spring of 2018. In the spring of 2018 the NRC conducted a QA inspection, covering Chapter 17.5 of the SSAR.

Community timeline. In 2018 the ACRS Committee met in May, August, October, and November to review selected SSAR sections, as shown on the slide. And today, as the slide illustrates, we're here for the final full Committee meeting.

TVA was asked to provide additional information associated with the approach to emergency preparedness. I would now like to introduce Archie to discuss the EP. Archie.

MS. MANOHARAN: Thank you, Ray. Good afternoon. Thank you for the opportunity to present today. As we mentioned I'm Archie Manoharan. I've been working in the nuclear industry for the last ten years, joined the licensing team at Clinch River in 2017.

And I would like to begin with the layout of the emergency preparedness approach in the application. To fully understand the emergency preparedness approach for Clinch River it's important to consider the information in three parts of the application.

Part 2, SSAR Section 13.3 in Section,

emergency preparedness, describes a dose based consequence or entered methodology for determining a plume exposure pathway EPZ for the site.

We have not selected a reactor design for the site. So, in this section the application is only seeking approval to use the methodology at a later stage, with design specific information, say in a COLA.

This methodology, along with the SMR design features is sort of the basis for the emergency preparedness approach described in the application. Based on the methodology Part 5 of the application has two distinct emergency plans.

Part 5 Alpha has major features of an emergency plan for a site boundary EPZ. And Part 5 Bravo has major features of an emergency plan for a two mile EPZ. Again, only major features are discussed in Part 5. There is no design specific information.

At a COLA, once the reactor design has been selected, and the dose based methodology that's described in 13.3 is adequately demonstrated, we would pick one of the emergency plans described in Part 5.

For example, if the selected reactor design meets the dose criteria at site boundary, we would go ahead and use Part 5 Alpha to create a integrated and complete emergency plan and COLA. If the reactor design meets the dose criteria at two mile EPZ, then Part 5 Bravo would be used.

The information in Part 5 meets the regulatory requirements if you consider it with the exemption requests described in Part 6. In Part 6 of the application two sets of exemption requests have been described, one to support the site boundary EPZ, and the other for the two mile.

Next slide. We're on Slide 17. And the dose based methodology described in Section 13.3 is consistent with the sizing rationale described in NUREG 0396. The NUREG introduced the concept of a generic EPZ, and recommends that a spectrum of accidents be addressed for the EPZ sizing.

So, consistent with that approach the methodology we are proposing in the application also describes, also addresses a spectrum of accidents. And more importantly, it has the same dose criteria for the plume exposure pathway EPZ as a recommendation in NUREG 0396, which is the one rem total effective dose equivalent, the early phase EPA PAG.

Consistent with the NUREG the technical criteria in the dose based methodology can be understood as Criteria Alpha, Bravo, and Charlie. Alpha can be understood as the plume exposure EPZ should be of, encompass of those areas where projected dose from design basis accidents could exceed the one rem TEDE.

Bravo the same, except for dose consequences from less sever core melt accidents. Criterion Charlie would verify that the plume exposure pathway EPZ is of sufficient size to provide for substantial reduction in early health effects in the case of more severe core melt accidents.

Next slide. So, we're on Slide 18. And this slide here describes the steps involved in implementing the methodology. The methodology at a high level contains four steps, starting with accident scenario selection. This is where you would rely on design and site specific information to do the appropriate accident selection.

For Criterion Alpha accidents you would rely on the bounding design basis accidents from Chapter 15 of the COLA. For the severe accident scenarios you would rely on the site and design specific PRA. And the criteria is actually shown here.

So, sequence. Firstly we'll start with

sequences with a mean core damage frequency greater than E to the negative eight per reactor year. And then you would further categorize them into criteria.

Bravo accident scenarios would be mean core damage frequency greater than E to the negative six, with intact containment.

And Charlie, the more severe core melt accidents, would be accidents with mean core damage frequency greater than E to the negative seven, or with containment bypass of the --

MEMBER KIRCHNER: Archie, may I interrupt here? So, I think this is mentioned in your application. For rhetorical purposes, if the design you choose, the PRA doesn't show any accidents greater, severe accidents. I'm looking at in particular greater than one E to the minus seven. Then I think you suggest putting in an alternate source term. Is that --

MS. MANOHARAN: That is correct. So, for Criterion Bravo, it's not listed on this slide, but there is an additional note in the methodology that even if you pick a reactor design that has no accident screened in for Criterion Bravo you still have to create alternate --

MEMBER KIRCHNER: Now, is -- Well, I'll

get a chance to ask the staff whether they're in agreement with this approach. But then, how would you come about, go about picking that source term?

MS. MANOHARAN: I think we can actually explain that during the example analysis.

MEMBER KIRCHNER: Okay. I'll wait.

MS. MANOHARAN: Which is in the next slide. Because we encountered that exact scenario in the example analysis. So, okay. So, moving on to the next slide, 19.

So, you would, after the steps one through -- I apologize. Can we go back to 18? Yes. So, after the accident selection, based on the cut off frequencies described here, Step 2 would be to determine the source term releases from the selected accidents.

Step 3 would be to calculate the dose resulting from these accidents at a distance from the plant. Four obviously would be to compare that to the EPA PAG limits to ensure that we are within that one rem limit. Next slide, please.

So, Criteria Alpha and Bravo, as I just mentioned, you would compare the dose calculated to one rem, and make sure you're not exceeding that. For Criterion Charlie, consistent with NUREG 0396 approach, you would calculate the distance at which the conditional probability to exceed 200 rem whole body exceeds in the negative three per reactor year.

CHAIRMAN CORRADINI: So, with that one, can you tell me how the one in a thousand is computed? I go back to 0396, and I'm lost. Tell me how that's computed. I understand the dose criteria. I don't understand what the frequency represents.

MS. MANOHARAN: Okay. I will bring in Alex to --

CHAIRMAN CORRADINI: If you want to do it later, that's fine. I, whenever it's suitable. I just want to understand what that is.

MS. MANOHARAN: We can do it now.

CHAIRMAN CORRADINI: Okay.

MR. YOUNG: So, my name's Alex Young. I'm working as a design engineer on the SMR project. Been here since September of 2014. So, the question is, you know, about the Criterion C dose criteria.

The conditional probably to exceed 200 rem whole body is one E minus three. So, we look at that. As you go out in distance from the release point, the reactor building, the probability of acquiring a certain dose goes down, based on meteorology.

So, we're looking at the distance at which the probability of acquiring the 200 rem whole body dose exceeds the one E minus 3.

CHAIRMAN CORRADINI: I got that part. I don't understand why -- So, let me, so, here's where I'm confused. I've now got accidents that fit in a range of greater than ten to the minus seven, but less than ten to the minus six. Yes, the frequency is one ten minus three.

So, have you subtracted a way, or taken out the initiating even frequency? This, the number sounds high to me, one in a thousand. I'm confused about one in ten to the seventh, ten to the minus seventh, versus ten to the minus three. That's where I'm struggling.

MR. YOUNG: Sure. So, for the Criterion C piece, on the previous line we kind of highlight the main CDF greater than one E minus seven per reactor year. So, that's looking at the probability of the event.

So, once you have the event, and you have a release, primarily based on meteorology statistics you have the probability changing as you go out in distance for that release. So, it's an additional factor in addition to the screening piece that's added in Criterion C.

MEMBER MARCH-LEUBA: So, in terms I can understand. Sometime we talk about a 500 year flood, 100 year flood. This is equivalent to that? So, you have the same, the initial source term. And now you consider the one thousand worst year that can possibly happen? Correct?

MR. YOUNG: Yes. That's a good analogy to categorize it.

CHAIRMAN CORRADINI: Okay. I'm still not there. Sorry. So, I've taken away the initiating event frequency, and all the estimates. And I've developed the source term. Then I release the source term, and I ask, what's the probability of getting a dose greater than 200 rem at a distance?

MR. YOUNG: You find out what distance it is at which the probability of getting that dose is one E minus --

MEMBER MARCH-LEUBA: But you run a thousand different years and pick the worst. Basically that's what you do, right? So, you start with a source term. And then, you propagate it, year one, year two, year three, using different winds, rain, different meteorological conditions, and pick the worst in a thousand.

MR. YOUNG: So, that's where the

meteorology comes into play, is looking at, you know, the meteorology that we have over time, how the statistics play out in that. What are the probabilities of having certain meteorological conditions that, you know, make it, you know, how that disburses.

CHAIRMAN CORRADINI: Okay. But if I might just jump in? So, the one in a thousand is due to the meteorology at the site? It's not due to the production of the source term?

MR. YOUNG: It's both. It's the combination. Because you have the initial even, which allows the probability of the release.

CHAIRMAN CORRADINI: No. That part I got. But once I get the source term, because it sits in this band between ten minus seven and ten to the minus six, now I have a source term. And the one in a thousand is just a meteorological uncertainty, or meteorological distribution?

MR. YOUNG: That's the additional factor that is applied to the propagation of the source term.

CHAIRMAN CORRADINI: Okay.

MEMBER REMPE: So, if I went to the next slide here, and I looked at that number. You call it a probability. But it's got a frequency unit.

MR. YOUNG: Yes.

MEMBER REMPE: So, wouldn't it be a probability? Doesn't have a unit -- Why does it have units of frequency if, I mean, earlier you called it a core damage frequency, something per reactor year.

Now you're calling this a conditional probability. Shouldn't it just be ten to the minus three, instead of per reactor year? This is kind of a basic question here. But I thought probabilities wouldn't be in per reactor year.

MR. YOUNG: Sure. So, we think, and a lot of times we think of, you know, probability. And we tie that to a frequency here. So, we're looking at, you know, the probability that you have that 200 rem dose at what distance for one E minus three per reactor year.

MEMBER SKILLMAN: Alex, I'd like to ask this. At least two times, and maybe three, you mentioned the coupling of the probability of the event with meteorology.

MR. YOUNG: Yes.

MEMBER SKILLMAN: And I'll just tell you, my background was Bellefonte. I was one of the original managers for, or B&W managers for Bellefonte. So, we got well-schooled in the Sequatchie anticline, and the Lake Guntersville, and the meteorology down in that section of Alabama.

But we were interacting with the teams that were doing the other TVA plants at Sequoya and Watts Bar, at Browns Ferry. And so, we got tuned into different meteorologies at different locations.

I understand you to say, if you look at the event frequency, and look at the meteorology, you then come up with a probability of someone getting dosed at 200 rem.

Does that say that if you put the plant at Clinch River it might have one probability? And if you put the plant at Sequoya or Watts Bar with a different meteorology, that will be a different?

Okay. Now, hold that thought. How do you predict that meteorology? Because it sounds to me like you're using a probability riddle for a natural event that, at least in my judgment is very variable. The uncertainty has to be huge.

MR. YOUNG: So, the meteorology that we used for this analysis, and for the additional pieces of this are based on data collected from the site, and analyzed over, you know, a period of time, in accordance with, you know, applicable regulatory guidance.

MEMBER SKILLMAN: Over what period of time?

MR. YOUNG: So, for SSAR Section 2.3, in accordance with regulatory guidance 1.23, that comes down to a minimum of two years of data.

MEMBER SKILLMAN: Why is two years sufficient for a siting decision, when that site will be employed potentially for 60 or 80 years?

MR. YOUNG: So, there are additional steps that continue to -- So, in addition with monitoring the site specific data over two years, you have to compare that to historical pieces as well, and different pieces in the area, to make sure that it's representative of the site, and over a period of time.

In addition to that, there's also on site monitoring that you continue to do over the life of the plant.

MEMBER SKILLMAN: Thank you.

MEMBER MARCH-LEUBA: That's scary. You're saying that I build my plant, I pay the money, and now I have to monitor the wind. And if the wind gets off outside you assume I lose my license?

MR. YOUNG: So, there's, to that question, what we're looking at is, we have changes in meteorology. We do a lot of analysis to, you know, show that that meteorology is consistent over a long period of time. And we include abundant margin within that meteorology to account for potential changes like that.

MEMBER MARCH-LEUBA: So, you're hoping that your monitoring is large enough that you'll never get caught?

MR. YOUNG: Absolutely. Yes.

MEMBER MARCH-LEUBA: But you are running the risk?

MR. YOUNG: That's an operational risk we take.

MS. MANOHARAN: Okay. So --

CHAIRMAN CORRADINI: So, let me summarize, since I started this. I want to make sure I am clear. So, the one in a thousand is based on the site meteorology, conditional on the fact that I've had a severe accident of a certain frequency band. And is it all those accidents that, and you look for the worst source term of that grouping of accidents?

MR. YOUNG: So, that comes down to the step of, you know, determine source term releases from selected accidents in determining the selected accident, that appropriate evaluation. So, you know, as you go through this you'll come up with the, for the accidents that screen in you would come up with the, you know, the bounding accident --

CHAIRMAN CORRADINI: Okay.

MR. YOUNG: -- evaluation.

CHAIRMAN CORRADINI: So, you're looking for the bounding source term within that frequency band. You then do the computation on some sort of weather variability. And the weather variability is what essentially the term is a one in a thousand? I want to make sure I'm clear. Have I said it correctly?

MR. YOUNG: Yes. The, yes.

MEMBER RICCARDELLA: So then, are we really talking like probability to ten to the minus nine? Yes. If we have a event probability of ten to the minus six, and then the, if that event occurs the probability of achieving this dose is --

MR. YOUNG: Yes.

MEMBER RICCARDELLA: -- ten to the minus third. So, we're talking ten to the minus ninth?

PARTICIPANT: No.

MR. YOUNG: The essential. So, you would have the even probability, which would be greater than one E minus 7.

MEMBER RICCARDELLA: Yes. Somewhere

between seven, six and seven.

MR. YOUNG: And then you would apply the factor to it, based on meteorology. And if the total frequency of the 200 rem dose exceeds one E minus -- It has to be, at that distance you have to be within a probability of one E minus three for the 200 rem.

MEMBER RICCARDELLA: So --

CHAIRMAN CORRADINI: You said it now.

MEMBER RICCARDELLA: So then, the real probability of that occurring, of that event occurring, and a person getting that dose is ten is to the minus nine, or somewhere between ten to the minus tenth and ten to the minus ninth, right?

MR. YOUNG: Yes. You'd have to have the probability of the event --

MEMBER RICCARDELLA: Yes.

MR. YOUNG: -- first.

MEMBER RICCARDELLA: Yes.

MR. YOUNG: And actually --

MEMBER MARCH-LEUBA: You would have to integrate --

MEMBER RICCARDELLA: It's a condition. Yes.

MEMBER MARCH-LEUBA: -- year one through 1,000 what the consequences are. So, it's not ten to the minus nine. It's much, much higher.

MEMBER RICCARDELLA: Why?

MEMBER MARCH-LEUBA: Well, because --

MEMBER RICCARDELLA: Multiple events.

MEMBER MARCH-LEUBA: This is the 1,000 year methodology. You can have the 500 year methodology, the 100 year methodology. All of those give you those. So, you have to do the interval of all of those to get that average. It's math.

MEMBER RICCARDELLA: But regardless, that's a conditional probability, right? So, that only applies if you have the event.

MEMBER MARCH-LEUBA: Ten to minus seven you're giving with.

MEMBER RICCARDELLA: Yes.

MEMBER MARCH-LEUBA: Because that's when you have the event.

MEMBER RICCARDELLA: Yes.

MEMBER MARCH-LEUBA: Now you're picking the worst possible year in a 1,000 --

MEMBER RICCARDELLA: Yes.

MEMBER MARCH-LEUBA: -- to propagate it to the end of EPZ. But if you had a better way you will still propagate some dose.

CHAIRMAN CORRADINI: But a lower dose.

MEMBER MARCH-LEUBA: It will be a little lower dose with higher probability. So, you will have to do some kind of interval. And I don't know how to write it out right now.

MEMBER RICCARDELLA: I'll have Dennis, I'll ask Dennis to explain it to me after the meeting.

MS. MANOHARAN: So, back on this slide, this is the example analysis that was conducted as a result of the staff's RAI. So, we use the NuScale design at Clinch River site to do a demonstration, an example demonstration, to show what the dose at site boundary would result from the NuScale design.

So, as you can see for Criterion Alpha and Bravo the doses are on, in that table. And they have significant margin to the one rem limit. And there's also additional margin built in within the calculation that resulted in that example analysis.

Moving on to next slide, Slide number 20. So as, both Dan and Ray had mentioned earlier, Part 5 of the application contains two major feature, two emergency plans, major features of emergency plan. One to support the site boundary EPZ, and the other for the two mile EPZ.

Now, what they do is they, both of the address the 16 planning standards of NUREG 0654. Once a reactor design is selected for COLA you would do the dose based methodology that Section 13.3 describes to pick your EPZ size.

So, if it is site boundary, then you go with 5 Alpha, and you would incorporate design specific information, and create a complete and integrated emergency plan.

If the dose is met at two miles you would take the Part 5 Bravo, incorporate the rest of the elements to make a complete and integrated emergency plan.

If for some reason you pick a reactor design that doesn't meet either site boundary or two mile, then we would have to come up with a new emergency plan and COLA. Next slide, please.

CHAIRMAN CORRADINI: Just one clarification. The thinking that you guys have come with, with this either or approach is, the two miles is bound to the EAB?

MS. MANOHARAN: So, the reason for two emergency plans is, when the plant parameter envelope was being developed at least one of them, we were confident that at least one design would meet site boundary EPZ. So, we pursued the site boundary emergency plan.

CHAIRMAN CORRADINI: But not all of them?

MS. MANOHARAN: We were confident that all of them would meet --

CHAIRMAN CORRADINI: Okay.

MS. MANOHARAN: -- two mile.

CHAIRMAN CORRADINI: Okay.

MS. MANOHARAN: Therefore, the two mile.

MR. STOUT: And two miles was a surrogate for scalable. You know, we, the staff had indicated through SECYs a willingness to consider scalable EPZ. We picked the number that we thought would bound all four designs, and be representative of scalable.

CHAIRMAN CORRADINI: Can I torture you one last time? So, did you do any sort -- Well, maybe I should ask the staff this. Somebody should ask someone this question, which is, if I did two years and had the appropriate meteorology, and then I looked back ten years, and I did the same thing, did I see a big difference in the, I'll call it the uncertainty, or the distribution function of the various types of meteorology. Was this done?

MR. YOUNG: So, you're asking, so, we did, we collected two years of onsite data. Did we look at how that compared to, you know, a longer period of time? Yes, we did.

We did comparisons, you know, from data that was collected from the breeder reactor project. We also did comparisons to operating fleets, or our operating fleet, data collected in surrounding stuff.

CHAIRMAN CORRADINI: Thank you.

MS. MANOHARAN: So, moving on to Slide number 21. Part 6 of the application describes the exemption requests that support the emergency preparedness approach in the application.

So, if you look at Part 6 there are two sets of exemption requests. One that support the side boundary EPZ, and one that support two mile. As Dan had mentioned, two mile is a surrogate for scalable.

And the only real exemption request we're asking for in two mile EPZ is to deviate from the ten mile. We understand that if we go with two mile then there would be a need for formal offsite emergency plans.

And for the site boundary, in addition to deviate from the ten mile EPZ, some, various elements of, let's say off site exercises and notifications, evacuation time estimate analysis, we're taking exemption, we're requesting exemptions from that.

MEMBER RAY: Excuse me. You said, if we go with two mile. And then I couldn't understand what you said after that.

MS. MANOHARAN: That there would be a need for formal off site emergency response plans. So, even if it is two mile, and not a ten mile, there would still need to be an off site response structure, if you will.

So, the site boundary EPZ is, let's say the most restrictive, and has the most number of exemption requests. And two mile is only asking to deviate from the size of the EPZ.

MEMBER RAY: Thank you.

MS. MANOHARAN: Next slide, please. So lastly, this is a summary slide that shows the emergency preparedness information in the ESPA, and how each of these pieces will be used in the COLA if at all the COLA is pursued.

So, in Section 13.3, as we've been discussing throughout this presentation, there's a dose based, consequence oriented methodology described. It's design neutral. It's not specific to any one particular design that informs the PPE. And we're asking approval of the methodology.

At COLA, once the reactor design has been selected, we would implement the methodology with design specific implementation, and figure out what the EPZ size for that particular reactor design at the site would be.

In Part 6 of the ESPA the set of exemption requests that have been requested. And those would be implemented based on the dose based methodology results at COLA. So, at COLA we would seek approval of a design specific plume exposure pathway EPZ size for the reactor design selected.

Lastly, the emergency plan, Part 5, two distinct major features of an emergency plan for site boundary and two mile are represented in the ESPA. At COLA, after the dose based methodology is implemented, the final EPZ size has been determined, we would pick the appropriate emergency plan.

It could be the site boundary in Part 5 Alpha, or Part 5 Bravo, or a new design, a new EP based on the reactor design. And we'll create a complete and integrated plan. And the next one? And that concludes our portion of the presentation. Thank you for the opportunity today.

MEMBER KIRCHNER: Thank you. I think what we, what you're hearing from us is, we, since we met last we've been struggling with understanding exactly how -- In NUREG 0396 they have a figure. It's, for the record I'll cite it.

It's Figure I-11, Page I-38, which is how that task force actually came to the recommendation for the ten mile EPZ for the larger fleet of reactors that existed. And this was shortly after WASH-1400, the Reactor Safety Study.

So, it appears to us that this is an integrated curve, as Member Rempe is pointing out. It's giving us probability, but not per reactor year. It takes a probability based on a conditional core melt of, on the order of ten to the minus five at the time.

And then, with that source term, propagates with, in this case they use straight line plume trajectories for the weather. And then we're able to come up with isoclines, so to speak, of dose versus distance.

So, that's the historical basis and background for the current ten mile. What has been puzzling us is, and what you're proposing, how you go through the calculation once you have a given, either a class of accidents that are severe, or even a dominant accident.

It's clear to us how you used meteorology to propagate dose. But it's not clear how this probability of ten to the minus third is arrived at.

So, perhaps that's a question we also take up with the staff. Okay. So, that's the concern. I hope I've summarized well enough why we're puzzling collectively here.

The methodology in principle makes sense. But we're, we have been puzzling over just why this is probability per reactor year. As Member Riccardella pointed out, a simplistic approach might be to multiply the two together and get very low numbers, not what are reason -- what are indicated as a fairly high number, one in a thousand.

CHAIRMAN CORRADINI: We're engineers. We want to get the mechanics right.

MEMBER KIRCHNER: Thank you.

MEMBER REMPE: Actually, again, because we were chatting, and trying to figure out what was going on, and I have not attended all your Subcommittee meetings.

But you mentioned at the beginning what if the, or someone asked, what if they don't even have a source term at something ten to the minus eight? They can't get something out. And you said that there was some sort of example you were going to show us.

And, was I distracted, and I missed what you were going to do if they have no source term, for example?

MS. MANOHARAN: So, what I was mentioning for that question is that for severe accidents if, what if you pick a reactor design that does not have a accident that screens in -- Can you go to one?

MEMBER REMPE: Yes. What if it's like ten to the minus ten?

CHAIRMAN CORRADINI: Screen it out.

MEMBER REMPE: Yes.

CHAIRMAN CORRADINI: Then it's not there.

MEMBER REMPE: If you totally --

MS. MANOHARAN: Yes.

MEMBER REMPE: And what, you're going to force them, you said earlier, to come up with something.

MS. MANOHARAN: Yes. So, there is an additional --

(Off microphone comments.)

MS. MANOHARAN: Yes. So, for Criterion Bravo there is an additional note in our methodology that says that even if there are accidents that are, that screen in based on your reactor design, you still have to create a source term, an alternative source term to analyze the severe accident. So --

MEMBER REMPE: And again, I haven't

attended your Subcommittee meeting, but remind me what are you going to do for how they created? Or that's to be determined, on how they're going to generate something?

MS. MANOHARAN: So, I think Alex can speak a little bit on that. But I will say, for example -- let's go to the next one. Sorry to keep jumping. So, this is the NuScale example, as I was mentioning. It's just an example to show how the methodology would be implemented.

So, Criterion Alpha would be the design basis accidents from NuScale's Chapter 16 analysis. And then Bravo would be the severe, less severe core melt accidents.

So, I will walk through the example, and what accidents screened in, and why it would make sense to have an alternative source. So, if we pick a reactor design that doesn't have screening.

MEMBER REMPE: So, with your example, which you claim is associated with NuScale, they generated an alternate source term? And that's really beyond your methodology. You don't know how they did it. They just came up with something that was their alternate source term?

MS. MANOHARAN: Not quite. So, they

don't, for example, the accidents that resulted in this example analysis for Criterion A would be the design based accidents, which is a combination of what, their LOCA and other accidents.

So, it's not just the design based accidents. So, it's more representative of their Criteria Bravo also. And then, Bravo was their most probable accident, which is the loss of DC power sequence, the most probably accident.

MEMBER REMPE: So, they didn't have to go to some -- Or did they tell you what the frequency was for those type of events?

MS. MANOHARAN: I think we know the answer.

MR. YOUNG: So, we do know what the frequencies are associated with those sequences that informed their design basis accident analysis. But those are proprietary to NuScale.

MEMBER REMPE: Okay. So, let's ask it in a way that -- They basically picked something below ten to the minus eight, and they went ahead and moved it up.

So, basically you're kind of forcing them. So, I'm glad I brought this up, even though I may have missed some of the details.

MS. MANOHARAN: You may, yes.

MEMBER REMPE: But they basically agreed to just take a hit --

MS. MANOHARAN: It's several magnitudes lower. So --

MEMBER REMPE: Yes. So, they basically agreed to take a hit, just so that they could do something.

MS. MANOHARAN: Because of the note in our methodology that you have to do the analysis.

MEMBER REMPE: Okay. And they were okay with that? Okay. Thank you.

MS. MANOHARAN: And that information, I do want to just, that information is in an RAI response to the staff. So, the staff has seen that analysis.

MEMBER REMPE: Thank you.

MEMBER RICCARDELLA: Could I ask why Row B has a higher dose, site boundary dose, than Row A? And it's less severe?

MR. YOUNG: So, Row A is based on NuScale's Chapter 15 design basis accident analysis. And Criterion A, the accident or sequences that were evaluated for that are based on several accidents, you know, happening. Criterion B is just looking at one of those accident sequences that informs A, which is a --

MEMBER RICCARDELLA: Okay.

MR. YOUNG: It's a more severe accident.

MEMBER BLEY: Design basis accidents aren't core melt accidents.

MEMBER RICCARDELLA: Okay.

MEMBER BLEY: Are not. So, the next one is more severe.

MEMBER KIRCHNER: Okay. Well, at this point then, if there are no further questions of the applicant from the members at this point? Okay. Well, let's change then to your team. Andy, please.

(Pause.)

CHAIRMAN CORRADINI: Okay. Mallecia, Allen? Who's going to lead off?

MR. FETTER: I'm going to start. Just getting us started here.

(Off microphone comments.)

MR. FETTER: Okay. Just my screen looked a little different. So, I was a little confused there. Good afternoon. I'm Allen Fetter. Mallecia Sutton and I are the safety project managers for the Clinch River nuclear site, early site permit application.

And I will be presenting an overview of

the staff's findings and recommendations, which were discussed at the four previous ACRS Subcommittee meetings. The technical reviewers are also here to address questions in their technical areas that, any questions you have during the presentation.

TVA submitted an early site permit application for the Clinch River nuclear site on May 26, 2016. The application was accepted for detailed technical review and docketing on December 30th, 2016.

TVA requested a permit approval for a 20 year term, along with approval for a plume exposure pathway, or PEP, emergency planning zone, sizing methodology, two major features, on site emergency plans and exemption requests for site boundary and two mile PEP EPZs. The plant perimeter envelope was based on four small modular reactor designs.

A staff overview presentation to ACRS on the Clinch River ESP was given on November 15th, 2017. The NRC staff's safety review of the application included execution of five audits, and one inspection, and issuance of 12 RAIs, comprising 50 questions.

The staff completed all advanced safety evaluations with no open items, and presented their findings at four ACRS Subcommittee meetings between May 15th, 2018 and November 14th, 2018. The advanced safety evaluations include 42 COL action items and eight permit conditions.

Staff cooperated with the U.S. Army Corps of Engineers, consulted with the Federal Emergency Management Agency, and engaged with the Department of Energy, the Tennessee Department of Environment and Conservation, and the U.S. Geological Survey, and the Tennessee Emergency Management Agency.

So, an early site permit plant parameter envelope values can bound a variety of reactor technologies, rather than one specific technology, an amalgam of values representing a surrogate nuclear plant.

The PPE values are bounding criteria used by staff to determine the suitability of an ESP site for construction and operation of a nuclear plant.

In the combined license application, when a specific technology is identified the PPE values are compared to those of the selected technology.

If design parameters of the selected technology exceed bounding ESP PPE values additional reviews are conducted to ensure that the site remains suitable from a safety and environmental standpoint for the construction and operation of the selected nuclear plant technology.

MEMBER KIRCHNER: Allen?

MR. FETTER: Yes.

MEMBER KIRCHNER: I'm going to interrupt at this point. I was going to ask this later in your presentation. Maybe I'll just put this down. And maybe you can address it later.

The, one of your permit conditions that you're going to share with us is the use of the Table 13.3-1, which is the PPE set of source terms by isotopes. And what if there's a variance in that?

Or are you confident that, maybe it's a question of the applicant as well, that if something in the fuel cycle that is used, we know that they're using LWR derivative fuel in most of the concepts that are under consideration.

But what if there's a variance in that table, that they exceeded one of these radionuclide amounts with the concept that they chose to go forward with, that COL point? What happens then?

MS. SUTTON: So, during the exemption and presentation Michelle will discuss that --

MEMBER KIRCHNER: Okay.

MS. SUTTON: -- in more detail.

MEMBER KIRCHNER: Excellent. Okay.

MS. SUTTON: Thank you.

MEMBER KIRCHNER: Thank you.

MS. SUTTON: You're welcome.

MR. FETTER: Okay. As stated before, the plant parameter envelope is based on four modular reactor designs, mPower, NuScale, Holtec, and Westinghouse. TVA's PPA is based on construction and operation of two or more SMRs at the Clinch River nuclear site, with a generating capacity of 2,420 megawatts thermal, or 800 megawatts electric.

Okay. This slide is for ACRS records. It depicts all of the advanced safety evaluations, and their associated accession numbers in ADAMS, that were provided for all the ACRS Subcommittee meetings.

MEMBER RAY: There's no assumption at this point as to the number of units that might be affected by any of the events described, right? It could be one. It could be all. Is that correct?

MS. SUTTON: So, during the exemption presentation -- This is just a overview of the staff's safety evaluation. We will address all those in details for you. I promise. So, hold that thought.

MEMBER RAY: So, I will indeed. But so, we're going to find, the answer is different for each of these. Is that what I just heard you say?

MS. SUTTON: No. That's not what I said,

sir. So, Michelle, do you want to --

(Off microphone comments.)

MS. SUTTON: Okay. Go ahead. Ask the question one more time.

MEMBER RAY: Is there any assumption in what we're reviewing here, in a multi module site, that only one of the modules will be affected at a time? I'm looking at events here that include vibratory ground motion, for example.

MS. SUTTON: Does any of the staff like to address the question?

MR. CAMPBELL: Well, let me address that. This is a plant parameter envelope for an ESP. There are a variety of assumptions that are put in by both the applicant and the staff in its review.

And with that said, we're looking at a number of different scenarios within that plant parameter envelope. So, that's how it's developed. And the plant parameter envelope encompasses all the designs. So, I don't know if TVA wants to specifically --

MR. FETTER: It looks like Alex --

MR. CAMPBELL: -- address that question.

MR. FETTER: -- wants to come to the --

MEMBER RAY: Before they do, let me just

say I would interpret what you just said to be that no, there's no assumption in what we're reviewing now that only one module would be affected by an event. That's how I interpret what you just said.

MR. CAMPBELL: In some scenarios that go into the plant parameter envelope, and someone who's actually an expert in this can correct me if I'm wrong.

There are scenarios where there's one module. There are scenarios where there's more than one module, if it makes sense. And, you know, the frequency of occurrence of more than one module is within that range that should be considered.

MEMBER RAY: Okay. Well, I think that you've answered the question. I'm at least going to understand it to be that we aren't limiting consideration to only a single module being affected in what we're discussing now. But that's my understanding of what you just said.

MR. CAMPBELL: And we'll confirm that.

MEMBER RAY: Thank you.

MR. CAMPBELL: Okay.

MEMBER REMPE: Actually, there was a guy from TVA who might be able to confirm it now.

MR. CAMPBELL: Yes.

MEMBER REMPE: And since we're doing our letter in the next --

MR. CAMPBELL: Yes.

MEMBER REMPE: -- few hours --

MEMBER RAY: Thank you.

MEMBER REMPE: Yes. I'd like to hear --

MEMBER RAY: Yes.

MEMBER REMPE: -- his response --

MEMBER RAY: We would too.

MEMBER REMPE: -- to my --

MR. YOUNG: Sure. So, my name's Alex Young. So, the question was about multi module accidents for the ESPA. Currently the way we've assessed the ESPA, based on the plant parameter envelope, the inputs that we have do not assume any multi module accidents. They're all based on single unit accidents, or single units events.

At the COLA stage, depending on the design selected, that's something that would have to be evaluated based on the design. But currently the assumption for the ESPA is only single module events.

MEMBER RAY: But what's the basis for that?

MR. YOUNG: The basis for that is based on the design information that we have available at the time for input into the ESPA for the PPE. We don't project, or believe that there are going to be multi module events that have to be considered for this.

MEMBER RAY: But that's a belief, as you express it, that I don't understand the basis for.

MEMBER RICCARDELLA: But if your plant parameter -- Excuse me, Hal, I'm sorry. If your plant parameter envelope is based on 800 megawatts, then doesn't that automatically address, doesn't that automatically cover multi-unit accidents for the smaller module, for the smaller units?

MEMBER RAY: Well, I don't know if you're asking me or not --

MEMBER RICCARDELLA: No. I'm asking TVA.

MR. YOUNG: So, part of the piece here is design basis accidents versus beyond design basis accidents. So, there's Chapter 15 analysis, design basis accidents, which for the information we have right now doesn't consider those multi module accidents, based on the design information that we have currently, when we developed this.

For the EPZ portion, you know, we have to consider those multi module accidents. And at COLA we still have to, you know, go and consider the possibility of those multi module accidents for

Chapter 15 as well. So --

MEMBER RAY: Yes. I understand it at the COLA period. Whether or not the early site permit parameters fit within what is then being proposed of the COLA is one of the issues that is, necessarily has to be addressed at that time.

But I guess it's just, you said, based on your understanding of the plants, this -- What's being described here isn't just a early site permit boundary based on a limiting size accident. You're actually talking about multiple units.

And now you're saying that the assumption is based on an understanding which is not part of this process. That only one of them at a time will be affected. And I just want to be clear that that's what's going on.

MR. CAMPBELL: At the stage of the COL the applicant would have to, with the specific design. Because the applicant here for the ESP looked at a range of different designs.

At the COL stage, when you have a specific design, then you can do that type of analysis, and establish what that is. And if that exceeds the parameter, then they would have to take a deviation or exemption.

MEMBER REMPE: So --

MEMBER RAY: Although the applicant did what you described, it doesn't sound like we did what you describe.

MR. CAMPBELL: our review is based upon the ESP, not on what will be done at the COL stage.

MEMBER RAY: I guess I'm asking, why is it, why are we even talking about multiple units, only one of which has an accident at a time? Why is that part of the discussion at this point?

I mean, I understand establishing an ESP. I don't understand talking about multiple units, only one of which is assumed to have an accident at a time, based on information that isn't part of this application.

I mean, I know that when the COL comes up this can be addressed. I grant that. But I don't understand why we're doing what we're doing at this point, relative to limiting the assumed event to one of several units that will be at the site that we're talking about.

And with that, I guess we ought to just leave it there and move on. I just don't understand it. At least we ought to be clear that that's what's happening.

MEMBER REMPE: If we could have TVA come back up for a second to the mic? I had a question, and I didn't get to get it in in the discussion. Okay. So, you did, as we talked about, you came up with some alternate source term based on a hypothetical

What if you learn more about one of these plants, and they determine that multiple modules are involved. How do you think that would affect your process you've developed here?

MR. YOUNG: So, our process does, you know, this is specifically talking about the EPZ methodology. So, that methodology does require us to look at those multi module events. And we'll look at those.

In our example analysis, for instance, for one of the events that we did look at as we were going through the screening criteria for Criterion C, one of those was considered a multi module event. It's a beyond design basis event. But it was required to be considered based on our methodology in the initial screening.

The second screening portion, so we have the E to the minus eight screening, and then we have the second screening for, at a greater frequency. It was then excluded from that, because it was only a single event. So, it didn't meet the second screening criteria.

MEMBER REMPE: So, basically if you, if they learn something new about their plant, and not picking on any particular one, and they decide suddenly, well, both modules or all 12 modules are going to be impacted by an event at a much higher frequency, your process could accommodate it?

MR. YOUNG: Yes. We have to consider that, yes.

MEMBER REMPE: Okay.

MEMBER SKILLMAN: Can you accommodate it without shopping for new meteorology? Yes. I'm pulling your leg. But I'm serious on the question.

MR. YOUNG: So, from what we know about the example analysis we considered, we would be able to meet our Criterion C dose requirements based on that, you know, assumed, if we assume that that multi module accident screened in. That would meet the dose criteria. It depends on the accident that would screen in. So --

MEMBER SKILLMAN: Okay. Thank you.

MEMBER RAY: Well, I just want to talk to the NRC at this point, not to the applicant. I just think it needs to be really clear what our understanding is of what the applicant is assuming in connection with this ESP. Because it was not something that I thought was explicit or clear at all.

MEMBER KIRCHNER: If I might summarize my understanding at this point? It's going in, the applicant has bounded the source term up to a single unit of 800 megawatts thermal.

And they've deferred on the multi-unit, say common cause, common mode failure kind of concerns until the COLA application, the COL application. And a PRA that would then have to be examined to see whether a multi-unit failure of some kind, or accident sequence would then lead to a source term that would exceed what they're currently asking for, as an exemption for either the one mile or, not one mile, the site boundary or two mile boundary.

If they come in at that point, and don't screen out multi-unit failures, and find that the dose exceeds the envelope, then they are not going to be in a position to get this exemption. Of course --

MR. CAMPBELL: They would have to develop additional information at the COL stage to demonstrate what that boundary, what the size of the EPZ would be, given those considerations. There's a full blown PRA done at the COL stage for a specific design.

Part of the issue here is, we're not approving results on the basis of only one specific design. What we're approving is a methodology. As I said in my opening, this is an approval of a methodology that can then be applied at the COL stage.

And the exemptions are to the, essentially the requirements with respect to the ten mile EPZ. That doesn't mean we're automatically approving either a site boundary or a two mile EPZ for a COL applicant. They have to make their case.

MEMBER RAY: Well, it's, I'm not sure that the issue of multi-unit failure isn't going to be addressed through the DCD, much less, not necessarily in the COL stage. But in any event, all I'm trying to do is figure out why, what we're assuming, and why we're assuming it. So that it's clear.

MEMBER DIMITRIJEVIC: Can we go to Slide number 5? Because it will be clear what we are asking. Because it says in that, PPEs based on construction and operation of two or more SMRs at the Clinch River site.

MEMBER RAY: Where, what are you saying, Vesna? I'm sorry.

MEMBER DIMITRIJEVIC: That in the last

paragraph, says that this PPE, the plan parameter envelope is based on construction and operation of two or more SMRs.

MEMBER RAY: Yes. So let's --

MEMBER DIMITRIJEVIC: So, why are we talking two or more?

MEMBER RAY: That's why I'm asking the question is, whether or not we assumed only one of these, or more, suffered a release that, is what we're talking about here in setting a boundary. And if we only assumed one, why?

(Off microphone comments.)

MS. HART: All right. This is Michelle Hart, from the staff. I didn't do the Chapter 15 analysis. But I understand the Chapter 15 analysis.

So, in general terms, the plan parameter envelope is developed based on current information, and does include consideration of one unit at a time, because we are, there's a presumption that GDCs 2, 4, and 5 will be complied with, so that you won't have common cause failures.

That you won't have, you know, much like you don't look at siting for more than one unit, at the currently operating plants we thought that that would also apply to a multi module site, until told differently from the specific design.

MEMBER RAY: Well, I understand. And I've operated a multi-unit site. An I know exactly what you're talking about. But it's also why I'm asking the question. Because it's not a resolved issue. And the only thing at the end of the day I'm seeking, is for us to be clear about what we're doing.

And I don't want anybody later to believe that what we have done here is agree that only a single unit in a multi-unit site need be assumed to fail. Notwithstanding multi-unit sites today that exist today elsewhere. I understand that very well.

MS. HART: I think the thing is that the information that we have, Chapter 15 was based on a non multi module unit. And so, the single unit was bounding --

MEMBER RAY: Exactly. That's right.

MS. HART: And so, it's, I hope it's clear that that's what we did. But there's no prevention of saying that if something came in for the COLA to use this ESP, if it doesn't fit within that PPE, whether it's a single unit or multi module event, that they would have to take a variance, and have to describe it more clearly.

MEMBER RAY: Yes. I mean, we think about

this in the DCD world also. And so, it's not just when a COLA comes in on an ESP, for a given site, I mean. Anyway, I think we've taken enough time here. Again, my goal isn't to try and change what's happened. I just wanted to be really clear about the basis for what I --

MEMBER RICCARDELLA: But isn't it fair to say we're approving a methodology to set the EPZ based on probabilities of various events? And when you get to either the DCD stage or for the COL stage, you're going to have a PRA that talks about the probability of single unit --

MEMBER KIRCHNER: And multi-unit.

MEMBER RICCARDELLA: -- and multi-unit events. And if any of those multi-unit events trigger these probability limits, they're going to have to be considered, right?

MEMBER RAY: Well, you're not going to be able to do that given the way the DCDs are envisioned today, as the design certification being approved. You're not going to have the information you're talking about.

MEMBER RICCARDELLA: Yes. So, maybe it's the COL stage. But at some stage you have to --

MEMBER KIRCHNER: I think it is the COL

stage.

MR. CAMPBELL: Yes. That is correct. It's at the COL stage when they have to do a full blown PRA.

MEMBER RICCARDELLA: Yes.

MR. CAMPBELL: If the frequencies of a multi-unit failure at a site are low enough that they don't have to be considered, they aren't considered. But if they're high enough, for a variety of reasons that may not be apparent at this stage, when we don't really have --

We have designs. But we have designs with, that really aren't solid, not necessarily approved at this point in time. In fact, we have no approved design at this point.

When you get to that, that's where you apply this detailed look at multi-unit failures that could exceed the cut off likelihood in terms of CDF. That's where this is done. It's done at the COL.

There are a lot of COL action items within an ESP that are simply saying, this is not an item we can make a decision on at this time, because we just simply don't have a design. We have a range of designs we're considering. And that's the way we've been doing ESPs now for five ESP permits so far. And we're in, we are consistent with that approach.

There are a lot of things that don't have all the information for at this time. But we have enough information to establish what the methodology is, and enough information to establish that one could come in with a design that might meet the site boundary, or two mile, or some other EPZ distance.

It might not be two miles. It might be three, or it might be one. But if it goes beyond the site boundary -- So, all of those things are covered in the COL.

And they're, I don't know the exact number from the SEs. But there are a large number of COL action items that we'll notify the COL applicant, you have to deal with this.

MEMBER RAY: You know, having sought an ESP I do understand and agree. What I was trying to understand is why we were going beyond what I think you said, to talk about multi module plants, implying, I thought, that we were only going to assume one of the modules have an event at a time.

And it was the additional small modular concept that I was questioning, not that the ESP goes beyond where it has traditionally gone in the past.

MEMBER BLEY: I kind of like everything

you said. What we're so, the one thing I would mention twice, you've said at the COL stage there's a full blown PRA. So far no COL applicant has performed a full blown PRA. They've deferred a lot of the detailed issues until just before fuel load.

MEMBER RICCARDELLA: Getting later and later. What do you do if you get to that stage and you say, woops. The zone has to be three miles, not two miles. That would probably be problematic.

MEMBER KIRCHNER: Harold, thank you. The clarity is needed. Let us address that when we deliberate over our letter on this matter and move on in the interest of time, Allen, if you could.

MR. FETTER: Yes. And in the interest of time I'm going to go over the next few slides rather quickly so that we can get to the staff's review of 13.3 and the exemption request. With that being said, if ACRS has any questions, please don't hesitate to interrupt.

Okay. For geography and demography, the staff review is based on information provided by the Applicant and the staff's independent confirmatory evaluation. Staff found that information to be acceptable. It meets the requirements of 10 C.F.R. 100.20.

For Section 2.2, nearby industrial transportation and military facilities, based on the information provided -- oops, we're still on that -- by the Applicant and staff's independent confirmatory evaluation, the staff found the information to be acceptable as information meets the guidance provided in NUREG 0800, Section 2.2.1 to 2.2.2.

Meteorology, discuss the site-specific information related to regional climatology, local meteorology, onsite meteorological monitoring, and long and short term atmospheric dispersion estimates.

As noted on the slide, site characteristics related to extreme weather were found to be acceptable for the Clinch River site. The onsite meteorological monitoring system was found to provide adequate data to represent the meteorological dispersing conditions at the site.

Site characteristics related to short term and long term atmospheric dispersion estimates were found to be acceptable. Based on information provided by the Applicant, the staff found all regulatory requirements to have been satisfied with no open items.

Okay, Slide 10, short term or accident atmospheric dispersion factors are X/Q. Estimates were developed for the exclusion area boundary and outer boundary of the Low Population Zone.

The exclusion area is defined in 10 C.F.R. 50.2 as that area surrounding the reactor in which the reactor licensee has the authority to determine all activities, including exclusion or removal of personnel and property from the area.

10 C.F.R. 50.2 also defines the Low Population Zone as the area immediately surrounding the exclusion area which contains residents, the total number and density of which are such that there is reasonable probability that the appropriate protective measures can be taken on their behalf in the event of a serious accident.

TVA used the NRC endorsed PAVAN Atmospheric Dispersion Model to estimate X/Q values for the zero to two-hour timeframe at the exclusion area boundary as well as the longer timeframes noted on the slide for the outer boundary of the Low Population Zone.

These X/Q values are intended to represent dispersion conditions that exceed no more than five percent of the time for the Clinch River site. The X/Q values, in conjunction with the estimated source term discussed in Chapter 15, are used to demonstrate compliance with 10 C.F.R. 5217 dose guidelines for design basis accidents.

Those dose guidelines include 25 rem TEDE for any individual located at the exclusion area boundary for two hours and 25 rem TEDE for any individual located at the outer boundary of the Low Population Zone for 30 days. I will now turn it over to Mallecia.

MS. SUTTON: For Slide 11, for Section 2.4, hydrologic engineering, TVA proposed adequate site characteristics and boundary design parameters for the inclusion in the early site permit. Design basis flood and maximum groundwater levels, and the accidental release, those estimates meet the regulatory requirements.

Staff concludes that the Applicant meets the early site permit regulatory requirements associated with the hydrologic engineering.

Slide 12, please. For geological site characterization, Section 2.5.1, vibratory ground motion, Section 2.5.2, surface deformation, Section 2.5.3, stability of subsurface materials and foundations, Section 2.5.4, stability of slopes, Section 2.5, based on evaluation of the information provided by the Applicant, and supplemented by knowledge gained through staff direct examination during site audits, the staff found Applicant adequately characterized the site in these topic areas in accordance with the applicable guidance.

Slide 13, please. Section 3.5.1.6, aircraft hazards, staff agrees with Applicant's conclusion that an aircraft crash probability is about an order of magnitude of ten to the negative seven per year or less and meets the provided NRC guidelines. Staff finds that the Applicant's approach is reasonable, and the probability value is acceptable.

Slide 14, please. So Chapter 11, radioactive waste management, Section 11.2.3 and 11.3.3, based on the staff's review of TVA's early site permit application, and subject to the staff's identifying several action items, the staff concludes that the normal plant permit, effluent source terms, and offsite dose meet the applicable regulatory requirements and are without undue risk to the public health and safety.

Slide 15, please. Chapter 15, accident analysis, staff evaluated the application and concluded that the Applicant's analysis meets the dose criteria specified in the PPE, includes a bounding accident release for the determination.

Slide 16, please. Section 17.5, quality assurance program description, staff evaluated the application and concluded that the Applicant's quality assurance program description for the Clinch River nuclear site ESP application meets the requirements of 10 C.F.R. Part 50, Appendix B, and 10 C.F.R. 50.17(a)(1).

Slide 17, please. Now that we have discussed all of the topic areas and their findings, the staff will now describe the evaluation emergency planning and related exemption requests. Recognize that TVA early site permit application was submitted in May --- in 2016.

This was before the staff started work on the small module reactor and other new technologies' rulemaking. According, the application and the review of the application by the staff is based on the current regulations and guidance.

TVA's early site permit application includes a methodology that, if approved in the early site permit, would be used in future combined license application and represents the specific merger reactor design and early site permit to determine the appropriate site-specific plume exposure pathway emergency planning zone size for the Clinch River nuclear site.

The submitted early site permit application requests exemption from certain emergency planning zone requirements if certain conditions are met. If these sorts of exemptions are approved as part of the early site permit, they will be accompanied by permit conditions specifying the circumstances under which these plans can be used in the combined license application.

If the exemptions are approved in the ESP, this Applicant can adopt these exemptions if it shows that its COLA PEP EPZ source term releases to the atmosphere are bounded by the non-design specific plant parameter source term information developed for the early site permit.

A future CO application featuring an SMR design, that fits within the plant parameter envelope established in the ESP, could apply the plume methodology to the design selected to determine the appropriate PEP EPZ size for the site and also demonstrate whether the conditions for either of the two sets of exemptions have been met.

The safety evaluation report for Chapter 13, Section 13.3 for the TVA Clinch River nuclear site --- early site plan application addresses the plans, design features, facilities, functions, and equipment necessary for the meteorological emergency planning that must be considered in an early site permit application that includes proposed major features of the emergency plans.

Now I'll turn the presentation over to Bruce and Michelle.

MR. MUSICO: Thank you. My name is Bruce Musico. I'm a senior emergency preparedness specialist. I and Michelle Hart reviewed the emergency planning information that TVA submitted in its ESP application.

The next two slides are a somewhat reduced version of the slides we presented before the subcommittee on August 22nd. And I refer you to the transcript from that day, because it provides more detailed explanation as well as answers to many of your questions from the subcommittee.

For emergency planning, the ESP application requested review of three key areas, and you're going to see an overlap with TVA's presentation as well, three key areas which consist of, first, the plume exposure pathway, the emergency planning zone sizing methodology, which Michelle Hart will discuss in detail shortly.

Secondly, the two major features, onsite emergency plans which were contained in Part 5 of the application, these include Part 5(a) which reflects a site boundary plume exposure pathway emergency planning zone, and Part 5(b) which reflects the two-mile EPZ, and it also includes the evacuation time estimate, or ETE.

The third review area was the 25 exemption requests that they provided. These include the two exemption requests which are applicable to both the site boundary and the two-mile plume exposure pathway emergency planning zone. And the remaining 23 exemption requests address portions of 10 C.F.R. 5047 (b), and Appendix E for offsite emergency planning related to the site boundary EPZ only.

Next slide, please. With regard to the 25 --- make sure I have the right slide --- with regard to the 25 exemption requests, the two exemption requests from 10 C.F.R. 50.33(g) and 50.47© would remove the ten-mile plume exposure pathway EPZ requirement. That same requirement is in both of those regulations.

The remaining 23 exemption requests, which are from 10 C.F.R. 50.47 and Appendix E to Part 50, would remove emergency planning requirements associated with offsite emergency planning. These requirements are associated with state and local emergency plans, public alert and notification, evacuation time estimate, and offsite exercises.

Next slide, please. This slide provides the basis for the staff's acceptance of the requested exemptions. The ESP application provides a basis for the establishment in the COLA of either a site boundary or two-mile plume exposure pathway emergency planning zone, and this is important, which maintains the same level of protection, that is dose savings in the event of a radiological emergency in the environs of the Clinch River site, as that which exists in the basis for a ten-mile plume exposure pathway EPZ, similar to what we used for the large light water reactors.

Next slide. This slide addresses the combined license application, or COLA. Upon issuance of the ESP the Applicant, TVA, acquires approval that is finality with conditions of the three key review areas that I just spoke of, first of all, the plume exposure pathway EPZ sizing methodology, the two major features emergency plan, the site boundary or the two-mile PEP EPZ, and the 25 requested exemption requests.

A COLA that incorporates, by reference,

the early site permit must identify the chosen SMR technology for the Clinch River Nuclear site and demonstrate that the EPZ sizing methodology supports either the site boundary or the two-mile plume exposure pathway emergency planning zone. The COLA must also provide a complete and integrated emergency plan.

For the two-mile plume exposure pathway EPZ, the COLA must provide both onsite and offsite emergency plans. For the site boundary plume exposure pathway EPZ, the COLA must provide an onsite emergency plan. And the COLA must also address all 16 of the COL action items and the four permit conditions. Those are 16 action items and four permit conditions associated with emergency planning.

Next slide, please. This slide addresses the EPZ size determination in the COLA. The determination of the EPZ size by the COL Applicant is required by two parts, two things, the COL action item, 13.3-1, and this particular action item reflects the language that was in the application Part 2 in Section 1333-14.

The COLA must identify the chosen SMR technology and the major features emergency plan, that'll be in the two-miles of the site boundary. It must provide detailed information that shows the ability of the small modular reactor to meet the chosen EPZ. And that would be utilized in the methodology. And the selected SMR technology must be the EPA early phase protective action guides.

Michelle Hart will address Permit Condition 1.

MS. HART: Hello again, my name is Michelle Hart. I'm a senior reactor engineer in the Office of New Reactors, the Radiation Protection and Accident Consequences Branch.

So for Permit Condition 1, this is related to, with the exemptions approved for the ESP, the COL Applicant can adopt the exemptions if it shows that the plume exposure pathway EPZ source term releases to the atmosphere are bounded by those in the non-design specific plant parameter source term information developed for the ESP. That's that table that's attached to Permit Condition 1, that's 13.3-1.

And as stated on the slide, the permit condition is that the Applicant would provide detailed information to demonstrate that the accident release source term information for the plume exposure pathway EPZ size determination analysis, using the selected SMR design, is bounded by the non-design specific plant parameter source term information used in the analysis supporting the exemption requests.

And that analysis would be done in accordance with COL Action Item 13.3-1 using the methodology in the SSAR, Chapter 13.3.

To go to your question, Dr. Kirchner, about what would happen if one of --- let's just say one of the isotopes is not less than the rest of the --- or the isotope in that table. My understanding is, because of the ministerial nature of the permit condition, if they cannot show that they are within that condition specifically, they may ask for an exemption, but they do not --- or a variance, but they do not automatically get to use the exemption requests that were approved in the ESP based on the condition with that design envelope, that source term information.

However, they may still be able to prove, through the use of the methodology, that although the source term is slightly different, or it may slightly exceed, that they still can prove that they have a site boundary or a two-mile emergency planning zone size according to the methodology.

MEMBER KIRCHNER: You've hit my question. Because it struck me, reviewing all the material, that it's almost --- you have to have agreement on the source term, of course. But the real thing you're regulating against is not the composition of the source term, it's the dose to the public.

MS. HART: That's correct. And there's some ---

MEMBER KIRCHNER: I had just worried that you might have an over-defined boundary value problem where ---

MS. HART: Right. In the subcommittee meeting, we did have a more full discussion of how they developed that source term. And I can discuss that again a little bit later if you would like. But they did add in a lot of uncertainty or a lot of margin to try to address that concern.

Next slide, please. So as TVA had told you earlier today that they ---

MEMBER REMPE: Michelle, can you ---

MS. HART: I'm sorry.

MEMBER REMPE: -- go back. I think I brought this up at the subcommittee meeting, but I can't remember how it was addressed. What if one of these designs happens to have a burp immediately after an event? And then something comes out starting on three and a half days, and it keeps going along. So what will you do if you see that kind of analysis? Or do they just get to stop after four days, and they don't have to keep it going?

MS. HART: Right.

MEMBER REMPE: And I've forgotten what your response was.

MS. HART: Right. Well, how I answered that at the subcommittee phase, and this is what I still believe, is that that's part of the implementation. And when we review their actual implementation, we will be looking at all the information that they have. And so if there is an issue there, it can be addressed.

What the permit condition non-design specific source term information is, is a 96-hour integrated. And so their release is longer than that, you know, we'll have to look at that when it comes in if there's --

MEMBER REMPE: And did I ---

MS. HART: -- some problem there.

MEMBER REMPE: -- mention that somewhere in whatever you --- again, I wasn't on the subcommittee itself, I just happened to be at that one meeting. And is that in your documentation somewhere, that you aren't allowed to just stop it at 96 hours?

You need to look for some sort of reduction or truncation of releases.

MS. HART: It is not specifically addressed. It's just the permit condition is written in such a manner, and we will have to say if your 96-hour integrated release does not meet that, that it would not meet the requirement to do the exemption.

MEMBER REMPE: Well, I can trust that you'll --- this will be adhered to even if you go on and get promoted to be a manager at a high level, that the staff will know to do that without any ---

MS. HART: That should be true, correct.

MEMBER REMPE: Cool, thank you.

MS. HART: Okay, Slide 22. So as TVA had mentioned earlier, they did have some technical criteria for developing their EPZ size methodology, that the plume exposure pathway EPZ should encompass those areas in which projected dose from design-basis accidents could exceed the EPA early phase PAGs.

The plume exposure pathway should also encompass those areas in which consequences of less severe core melt accidents could exceed the EPA early phase PAG, and that the plume exposure pathway EPZ should be of sufficient size to provide for substantial reduction in early health effects in the event of a more severe core melt accident.

Next slide, please. TVA did go through this earlier. I guess I probably don't need to repeat it in detail. But certainly the features of the EPZ size methodology are that they will select their accident scenarios, and that would include design-basis accidents, just taking that directly from the siting analysis that they do in Chapter 15.

And then you look at the severe accidents using the COLA site and design specific probabilistic risk assessment, should include all modes, internal and external events, applicable fuel handling, and spent fuel pool accidents, and also consider multi-module accident considerations.

And then you would categorize that in the two different categories, the more probable less severe core melt accidents with intact containment and then less probable, more severe, core melt accidents with either containment bypass or containment failure.

Once you categorize those accidents, you would determine the source term releases to atmosphere and its --- there's not a specific discussion as to whether you can do bounding or should do all of them. They can choose at that time. It's an implementation thing we would also evaluate.

So the source terms, there may be several scenarios in a different category, and they may determine to look at them all or they may categorize them and get us the bounding. When you calculate the dose consequences at distance from the plant, and then you compare those doses to the dose base criteria.

Next slide, please. So to go in a little bit more detail about the TVA dose-based plume exposure pathway EPZ size criteria, the quantity that we're looking at is the dose to an individual from exposure to the airborne plume during its passage and to groundshine using average atmospheric dispersion characteristics for the site.

And what we mean by average atmospheric dispersion characteristics for the site is not referring to the same analysis that was done in SSAR Chapter 2 and approved for the ESPS site characteristics. Instead, it's referring to evaluating the accident consequences using site-specific meteorological data to determine doses that are based on 50th percentile atmospheric dispersion factors.

And the staff expects that the Applicant may use the calculation tools that are used for a severe accident consequence analysis. For example, in the environmental report there's no specific tool identified in TVA's methodology.

But for example, the tool that is mostly used is the MACCS code, and so it can take a year's worth of hourly meteorological data. And you can run --- it can account for uncertainty in weather, including over the duration of the accident release.

It models atmospheric transport and dispersion by sampling one year of hourly weather data for the site, and it can model shifts in wind direction. It uses a Gaussian plume segment model, and so each plume segment, the start time and duration is chosen by the user. So it can be adjusted to the shape of the accident release, if that makes some sense.

We'll head in the wind direction and speed, as sampled from the site-specific data, and the start time of that sampling is random over the year. So therefore, two plume segments released at adjacent times may be traveling in different directions at different speeds the way that MACCS does the modeling.

In practice, when we're saying that they would look at the 50th percentile, or the mean doses, excuse me, in practice the analysis would run several weather trials with the same release source term for each weather trial but differing atmospheric dispersion and transport based on the sampling of the year's worth of data. And the resulting mean dose of our weather trials would be taken as the output.

Yes?

MEMBER DIMITRIJEVIC: I don't have a question, I just have a little correction, not for just slide, but there was a slide there, the airplane crashes where you have probability with the reactor. Every time when you have a pattern, it's not probability. Probability doesn't have a unit that's frequent. And you should change that throughout, because of the issue.

MS. HART: Thank you.

MEMBER REMPE: And if you agree with that statement, and hopefully, you'll help the TVA folks come to that conclusion too --

MS. HART: Yeah. And I think I understand what you're saying. And it's not something that I brought up with them before. So hopefully, we'll see what happens.

Okay, so for the rest of this slide, it reiterates the actual criteria that they have proposed, is that for design-basis accidents and more probable less severe accidents, those are the ones with intact containment, that the dose criterion is one rem, total effect of dose equivalent from a 96-hour exposure. And that is the lower end of the dose range of the EPA PAG for early phase protective action such as evacuation and sheltering.

And for the less probable, more severe accidents, and you see that I have repeated it, but they would calculate the distance at which the conditional probability to exceed 200 rem whole body from a 24-hour exposure exceeds ten to the minus three. And they did say per reactor year. The 200 rem is, of course, the acute dose at which radiation induced early health effects may begin to be noted. And so I've heard ---

MEMBER KIRCHNER: Once more, we belabored this earlier.

MS. HART: Yeah.

MEMBER KIRCHNER: But just so we're on the same page, this is an integral effect, this ten to the minus three?

MS. HART: I have to admit that this did not get practiced in the example calculation, because there was nothing that screened into that category. In general, what we are seeing from some of these small modular reactors, there's not very many accidents that may be in that category, if any at all.

So I don't know that you would have more than one source term affecting that evaluation. There may be, depending on the design. I think it's mostly going to be an effect of the weather.

And one of the things that we can do with this is up to the implementation phase, it's not discussed in their methodology or discussed in our evaluation. But in implementation, you know, MACCS runs one year at a time. But you can do more than one year by running another set of MACCS analyses.

And so if there's some concern or question, if you're not able to tell from the pre-processing of the weather, you know, to determine if you've got a bad year, or a worse year or, you know, from that perspective, if there's some need to have to do more than one year's worth of MACCS runs, then that is something that can be done. It would be evaluated based on the information that we have at the time of the implementation at the COL though.

MEMBER BLEY: Let me jump in here, Walt. I've been trying to catch up a little. But this deal about the ten to minus three, if you go back to 0396, and you go back to the figure, and Walt asked me about this last night, Figure 1-11, there's a curve for 200 rem. And what that curve says is, right at the site, only eight percent of core melts can get you 200 rem, even right at the site.

And by the time you get out to 20 miles, and these are results from WASH-1400 that got adapted for this report, when you get out to 20 miles, the curve's dropping off so fast that you hit only one in 1,000 core melts can have an effect on you. 0396 talked about, for severe core melt accidents, you ought to have a substantial reduction in health effects.

MS. HART: Right.

MEMBER BLEY: And nowhere does it say that substantial drop is ten to the minus three. But that's kind of what everybody is doing. And it's based on that one curve and then applying it to new reactors as well. I thought that worth throwing in.

MS. HART: Are there any further questions, concerns about that?

(No audible response.)

MS. HART: Okay. So next slide please. So the staff's review of TVA's proposed plume exposure pathway ETZ size methodology, we did compare the methodology and the dose criteria to the study used as the technical basis for the current regulatory requirement for a ten mile plume exposure pathway EPZ requirement, that is as we've been discussing NUREG 0396.

And the staff has determined that the features of TVA's methodology are consistent with the study that was done in NUREG 0396 in that it considered a range of accidents. It performs an accident consequence analysis and determines an area outside of which early protective actions are not likely to be necessary to protect the public from radiological releases.

And so therefore, the staff concludes that the Applicant's proposed methodology is reasonable and consistent with the analyses that form the technical basis for the current regulatory requirements of a plume exposure pathway EPZ of about ten miles in radius.

Next slide, please.

MEMBER BLEY: Michelle?

MS. HART: Yes?

MEMBER BLEY: For several reasons, I've been going through 0396 in great detail recently. This is one of them. Some 50 years ago, all the quantitative judgements in it were based on Wash 1400 which was, at that point, three or four years old.

Has anybody on the staff revisited 0396 and thought about it in light of what's been learned in the last 50 years?

MS. HART: I can let somebody from the Office of Nuclear Incident Response respond to that if they would like.

MEMBER BLEY: They must have run out the door.

MS. HART: Yeah. There're some folks here. I mean, certainly, we are going through the rulemaking for the emergency preparedness and for SMRs and other new technologies.

MEMBER BLEY: Still point at 0396. The logic there is great.

MS. HART: The logic is what we're using. Now, if you're asking have we re-evaluated it in the context of the currently operating reactors, I can't necessarily speak to that. And I don't know that that's what you're asking.

MEMBER BLEY: I think we're using more than the logic. I think we're using some of the quantitative information as well.

MS. HART: Well, I think certainly continuing to use the EPA PAGs for the early phase as the basis for how you determine EPZ size, we're still sticking with that idea.

MEMBER BLEY: Yeah, but we're picking the chance of what the dose is at some distance from very old information.

MS. HART: In TVA's methodology, yes, they did.

MEMBER BLEY: I didn't see anything in the rulemaking. I mean, there would be a change in that.

MS. HART: The rulemaking, as I recall, does not have that specific evaluation in the rule language itself.

MEMBER BLEY: That's true.

MS. HART: About the very severe accidents.

MR. SCOTT: I figured it out, thank you, with help. This is Mike Scott of the NCR staff. Talking to my colleagues here, we're not aware --- the question is is there a current effort ongoing to update 0396. Our answer is we're not aware of one. Was that the --

MEMBER BLEY: Thanks. I'm not either. And it just struck me, you know, it might be worth somebody doing that.

MR. SCOTT: It's an interesting question that we'll consider. Thank you.

MEMBER BLEY: Not that I'd hang it on this particular application. But I think it's about time we thought about it.

MS. HART: Okay. So then to Slide -- what is this, 26 -- so for the exemption requests to determine if --- to put a boundary around what we considered when we were looking at it in the ESP, since there is not a specific design included in this, TVA developed a non-design specific accident release source term that would meet the plume exposure pathway EPZ size criteria which are intended to be used as plant parameters for the purposes of the exemption request.

This source term is in Table 13.3-1. It is an isotopic total release activity over 96 hours which results in a Total Effective Dose Equivalent of about 0.9 rem at the site boundary. It's the same idea as the plant parameter envelope in general that's done for the ESP, specifically for the design basis accident source term. And it's intended to envelope an unknown design. And it's referenced in Permit Condition 1 for the adoption of the EP exemptions.

This non-design specific source term used information from two different designs from three accidents, two DBAs, and one severe accident. The two

SMRs were at the lower end of the range of the PPE and at the upper end of the range of the PPE as far as reactor thermal power.

And when they did this, they took the maximum activity that could be released in any time period from any of those three accidents from the two reactors. They added a 25 percent margin, and when they tried to back calculate from the 1 rem criterion, there was also some additional adjustment to some of the isotopic values. And then they calculated the final source term to result in some margin to the dose criterion, so about 0.9 rem at the site boundary.

And so it's the plant condition, plant parameters for the condition to use it for either the site boundary or the two-mile emergency planning zone. There's not a separate table for those two different distances.

And that concludes my portion of the presentation. I will turn it back over to Mallecia.

MS. SUTTON: The staff presented its review on findings on emergency planning for TVA Clinch River early site permit application. The staff concludes that the PEP EPZ size methodology is acceptable for determining the appropriate size of the PEP EPZ for the Clinch River nuclear site. Because it's consistent with analysis that formed the clinical basis for the current ten-mile PEP EPZ.

The two major features in emergency plans are acceptable, because they meet the applicable standards of 10 C.F.R. 5047 and requirements of Appendix E to 10 C.F.R. Part 50. If the exemptions are approved for the ESP, the Applicant can adopt exemptions if it shows that its COLA PEP EPZ source term release to the atmosphere is bounded by the non-design specific plant parameters source term information developed for the ESP.

The exemption requests are acceptable, because they are authorized by law, will not present an undue risk to the public health and safety, are consistent with the common defense and security, and special circumstances are present.

In previous subcommittee meetings, we have presented the staff's review and findings relative to this application for an early site permit at the Clinch River nuclear site. Today we presented an overview including more details in the emergency planning and related exemption request. The safety evaluation is complete with no open items.

The next step in the process is the mandatory hearing in front of the Commission in 2019.

The staff looks forward to an ACRS letter on the staff review. And this completes the staff presentation.

MEMBER KIRCHNER: Thank you, Mallecia.

MEMBER BLEY: I want to make just a ---

MEMBER KIRCHNER: Yes?

MEMBER BLEY: -- very minor comment which could --

MEMBER KIRCHNER: Go ahead, Dennis.

MEMBER BLEY: -- be editorial. In the licensee's report, Chapter 13, they go through the steps and the methodology. And they do that well, and they say find these scenarios, then group the scenarios by the kind of things that failed and what the consequences are. The next step should say for the groups, scenario groups, find the frequency. And it doesn't. It just says for the scenario. Just a comment for you.

MEMBER KIRCHNER: Other members, any questions of the staff while they're here in front of us? Then if not, we'll turn to the public.

(No audible response.)

MEMBER KIRCHNER: Okay, thank you again. Are there any members of the public in the room who wish to make a statement or a concern? Please step forward, identify yourself at the mic, and make your comment.

Seeing no one coming forward, is there anyone, member of the public, on our bridge line who wishes to make a comment? If so, state your name, please, and make your comment.

(No audible response.)

MEMBER KIRCHNER: Hearing none, at this point, Mr. Chairman, I'll turn it over to you.

CHAIRMAN CORRADINI: Thank you. So I'll thank members of TVA and the staff. And I think we're done with this subject. So we're going to take a short break, so we change out and talk about Seabrook next. So we'll be coming back at 3:15.

MEMBER BLEY: We're ahead of time. We can't start that until the scheduled time.

(Off the record comments.)

MEMBER BLEY: 2:30 or 2:45. I don't have my glasses.

(Off the record comments.)

MEMBER BLEY: That's 45 minutes, not 50.

(Off the record comments.)

(Laughter.)

CHAIRMAN CORRADINI: So once again, we'll see you in 15 minutes. Thank you all.

(Whereupon, the above-entitled matter went off the record at 2:59 p.m. and resumed at 3:13 p.m.)

CHAIRMAN CORRADINI: Okay, why don't we come back into session. Our next topic is going to be Seabrook, Unit 1, license renewal application. And I'll turn it over to Member Skillman.

MEMBER SKILLMAN: Yes sir, thank you, Mike. Ladies and gentlemen, this meeting this afternoon brings us to a very important time in Seabrook's life. We have been involved in license renewal of Seabrook since our meeting in 2012. It has been over six years. And in intervening time, from an original application, and then updates to the application and the safety evaluation, through years of work on Alkali-Silica Reaction, we come to today.

And so through the presentation and letter writing we will address both the license renewal application and Alkali-Silica Reaction. And with that opening comment, I will turn it over Joe Donoghue, please.

MR. DONOGHUE: Okay, good afternoon. Thank you, Chairman Corradini, and Mr. Skillman, and members of the ACRS full committee. I'm Joe Donoghue, I'm the deputy director of the Division of Materials and License Renewal in NRR.

We thank you for the opportunity given us to present the results of the staff's review of the license for an application for Seabrook Station, Unit 1. This review began many years ago, and as Mr. Skillman alluded to, one of the main technical issues that prolonged the review was the Alkali-Silica Reaction affecting concrete structures and then the licensee's development of methods. And I'll review those methods for managing the phenomenon.

On October 31st, the License Renewal Subcommittee of the ACRS heard detailed presentations from both the Applicant and the staff on ASR and the basis for closing out that one open item of the license renewal. On November 15th, the subcommittee heard from the Applicant and the staff on the closeout of the remaining open items in the SER.

Our presentation will be led by our project manager, Butch Burton, and other members of the staff and the management that are here, Dr. Allen Hiser, our senior technical advisor, Eric Oesterle, chief of the project's branch in our division, and there's other managers and other technical staff that contributed to the review that are present and that will support answering any questions you have.

We also have, I think, maybe on the phone, Region I staff who will provide inspection support and provided presentations to you during the subcommittee meetings.

Again, we look forward to answering any questions you have and having a full discussion. And I'll turn the presentation over at this point to the NextEra team and their regional vice president from the northern region, Mr. Eric McCartney.

MR. McCARTNEY: Thank you, Mr. Donoghue. Good afternoon. My name is Eric McCartney. I'm the regional vice president for NextEra Energy with responsibility for the Seabrook Station, Point Beach Station in Wisconsin, and the Duane Arnold Station in Iowa.

Today we're here to talk about the Seabrook Station. We appreciate the opportunity to come and provide our presentation of our license renewal application and all the work we've done over the last six years, as Mr. Skillman mentioned. And we look forward to a good discussion and answering any questions that the Committee may have about our program and our process.

We are committed to the safe, and reliable, and sustained operation of our nuclear fleet. And as we do that --- if you'll turn the slide, please --- there we go. This is our nuclear excellence model. And this provides the framework for how our fleet has operated since 2008. It's based on a set of core values and principles, and those have not changed since its inception, and they will not change.

And we use this as a road map of how we operate our fleet going forward. So I won't go through this as we've discussed this a number of times already. But this continues to be at the heart of how we manage our stations and our leadership model to drive safe, reliable, and sustainable operations of our fleet.

Today I have with me Mr. Mike Collins. He's our engineering director. Next to him is Mr. Ed Carley. Ed Carley's our license renewal supervisor. And next to Ed is Ken Browne. Ken Browne is our licensing manager. And then seated over here to my right is Rudy Gil. And Rudy Gil is our engineering program manager. And they will provide the technical responses to your questions today.

And with that, I'll turn the presentation over to Mr. Collins.

MR. BROWNE: Thank you, Eric. Good afternoon, Mr. Chairman. I'm Ken Browne, licensing manager for NextEra Seabrook. I've been at Seabrook for approximately 28 years, beginning in the Operations Department as a non-licensed operator, then as a licensed senior reactor operator working and controlling various positions, including shift manager and eventually director of operations.

I've also held the position of training manager of accredited programs and most recently as the licensing regulatory compliance manager and also the management sponsor for the Alkali-Silica Reaction project at Seabrook.

As we discussed at our ACRS Subcommittee meeting last month, this station has continued to engage in accumulating the best practices from the industry in developing our existing engineering programs as well as enhancing our aging management plans to ensure Seabrook is maintained to the highest safety and material standards.

Since we've been operating, NextEra Seabrook has always made it our highest priority to operate our facility with nuclear and public safety as the overriding focus in all that we do. Each of us that work there and live near the area recognize the location of our facility places a personal responsibility and accountability on all of us to protect the health and safety of the public surrounding Seabrook.

We also recognize the valuable resource that Seabrook represents and continues to provide for many years as a major proportion of safe, reliable, and clean energy in the New England area. We look forward to the Committee's questions. And I'm going to turn the panel over to Mike Collins, our engineering director, to guide us through the presentation, including some background on the station. Mike?

MR. COLLINS: Good afternoon. Again, my name is Mike Collins, Director of Engineering at Seabrook Station, 37 years in the industry, 17 years with Stone and Webster Engineering, with new build and continuing services, the last 20 years with NextEra Energy, Seabrook Station, five of which as engineering director.

(No audible response.)

MR. COLLINS: So our agenda for this afternoon, again, our introductions, I'll provide an overview of site and station description. Ed Carley will then review our license renewal application and our Aging Management Programs, review the safety evaluation report and closure of the previous open items. There'll be then closing remarks. And in summary, we'll end with NextEra Seabrook has met the requirements of 10 C.F.R. 54 for issuance of a renewed license for Seabrook Station, Unit 1.

Just so we won't bore the group, I've changed up the slide from previous. This is a picture of the station and some of the main structures, our intake, excuse me, discharge and intake structure, a circ water and service water pumphouse, certainly our containment enclosure building where the reactor building is housed within, our Unit 1 turbine building, fuel storage building, waste process building. And this area of the plant is our primary auxiliary building, our control building which houses our two emergency diesel generators.

As you know, the Atlantic Ocean is the normal heatsink for cooling at 100 percent power. We also have a standby cooling tower which is a seismic Cat 1 mechanical draft cooling tower which provides additional safe shutdown capability for the station.

Next slide, please. Plant status, recently completed our latest refueling outage, fueling outage OR-19, which we completed 10/29/18. Our next refueling outage at the end of Cycle 20 is spring 2020 in the April timeframe.

Our capacity factor for 15 of 19 cycles

has been greater than 94 percent with a lifetime capacity factor of 87 percent. As you can see with the listing of our cycle capacity factors, we've had an excellent operating history over the last cycle. Capacity factor performance is representative of our solid equipment reliability and our material condition for the station.

Next slide, please. In order to maintain high capacity factors, Seabrook continues to improve equipment reliability and material conditions of the station. Running down just through some items, for equipment reliability improvements, our main generator stator rewind, in the process replacing our vital batteries and our vital inverters, our generator step-up transformers replaced --- there's three of those that we fully replaced two outages ago.

As part of our Aging Management Program, our mechanical stress improvement process completed for all reactor vessel nozzles. Also Aging Management, we continue with our process of replacing all our above-ground service water piping with the high chroma AL6XN material. We've upgraded our incore detectors and have been aggressive with replacing our process control circuit cards and our solid state system circuit cards.

Two outages ago, we sent out our rod control motor and generator sets for refurbishment. And lastly, for all four reactor coolant pumps, we now have shutdown reactor coolant pump seals.

We are committed, NextEra Energy Seabrook, to maintain high levels of safety, reliability, and performance of our plant equipment.

DR. SCHULTZ: Mike, excuse me. You mentioned two of the items on the list that you attributed to the Aging Management Programs. And then you stopped listing what the remainder were for. You mentioned reliability and Aging Management halfway down the list. Is that the full characterization of why you made these changes?

MR. COLLINS: Yes, it is. With the ones that I didn't mention, Aging Management, those are driven by system engineer advocacy, trends of equipment such as the GSU. We're watching very closely the offgassing of the old generator step-up transformers. Those go in for our long term plant reliability plans. And we put them through the process, do the engineering, do the maintenance, and do the replacements either online or during the outages.

DR. SCHULTZ: So that's how you separate

them from what you would term an Aging Management Program improvement?

MR. COLLINS: That's correct.

DR. SCHULTZ: Okay, thank you.

MR. COLLINS: Thank you. At this time, I'll turn the program over to Ed Carley to start our discussions on our license renewal application.

MR. CARLEY: Good afternoon. I'm Ed Carley. I am a 35-year veteran of Seabrook Station and been in various organizations, quality assurance, licensing, engineering projects. In 2008 I joined the team developing the license renewal application as the time limit aging analysis lead and the environmental lead. And shortly after, the application was submitted in 2010. I also took on the role as licensing lead.

For the last four years, I've been the project manager for the license renewal application and resolution of ASR and the current licensing basis for concrete affected structures.

The original license renewal application was prepared onsite by Seabrook Station personnel. The team was supplemented by staff and contractors with various experience in license renewal and those that were former plant employees that were familiar with the history of the plant. Our corporate fleet also provided experienced personnel in license renewal and provided oversight to the project.

Our application was prepared in accordance with the standard review plan that's listed up here, followed the standard format for an application. We filed the guidance of NEI 95-10. And we developed our Aging Managing Programs in accordance with NUREG-1801, commonly referred to as GALL. Our initial application was submitted as GALL Rev 1.

Since that time of submittal, we have performed over 65 updates, some of those were proactive, some were related to REIs, and also produced eight annual updates to keep the application current.

We've addressed all ISGs that have been issued since the initial application was submitted. And we have performed a consistent review to GALL Rev 1 and GALL Rev 2, and provided updates to our program where we felt necessary to come in compliance with GALL Rev 2 for those programs.

Next slide, please. This is a table of our relationship in the final SER to the GALL. Fifteen of our programs we consider new. We consider 29 that were existing. We do have six plant-specific programs to which we have discussed in very much detail being the ASR and Building Deformation Programs.

Next slide, please. In relation to the Safety Evaluation Report that was issued by the staff on September 28th, it documented no open items and no confirmatory items. There were seven open items in the previous SER in 2012 as discussed earlier which are listed here. The first six we did discuss on November 15th. Of those, the programs for treated borated water, operating experience, and part of the Steam Generator Tube Integrity Program were resolved by adoption and incorporation of the ISG guidance that was applicable to those programs.

The other portion of the steam generator tube integrity and the pressure temperature limit open item were addressed by licensing actions in Part 50 for license amendments that changed our operating license to resolve the open item. Pressure-temperature limits have been approved out to 55 effective full power years which will take us through the period of extended operation.

Of the remaining open items that we haven't discussed, Bolting Integrity Program was related to a seal cap enclosure that was placed on a safety injector valve that was leaking during one of our operating cycles. We have, following cycle outage, we removed the seal cap enclosure, replaced the valve to eliminate the leakage of that valve.

In relationship to the IWE program, this was in relationship to the water that had accumulated in our annulus area. And there was a concern that we may have had degradation against our liner. To resolve that issue, we have established a weekly PM that verifies that area is in a dewatered state.

We have performed UT measurements around the liner at the area of the moisture barrier, confirmed there is no degradation of the liner in those areas. And we also have a commitment to perform that UT every five years, excuse me, every five cycles.

And the last item, which is the Structures Monitoring Program, we discussed quite extensively on October 30th -- can I have the next slide there -- and this is related to our Structures Monitoring Program. Structures Monitoring Program was developed in accordance with the GALL. However, because of ASR and building deformation, it is now augmented by supplemental plant-specific Aging Management Programs, one for ASR and one for building deformation.

As we discussed use of this flow chart earlier, the structural capacity that came out of the large scale test program at the University of Texas, dose limits have been incorporated, that keep us bound by that testing program, have been incorporated in the Structures Monitoring Program.

And also, the structural demand portion where we have performed -- in our performing analysis of our seismic category Cat 1 structures, those parameters to maintain us within the bounds of those evaluations area also incorporated into the Structures Monitoring Program. Frequencies, limits, and trending are performed in accordance with the Structures Monitoring Program to verify that we will not exceed the limits prior to reaching the next inspection interval.

MEMBER REMPE: Excuse me.

MEMBER MARCH-LEUBA: Go ahead.

MEMBER REMPE: Just to make sure that we have the facts correct, because we've seen some different states, I believe, and so just confirm for me, that you first detected visual indications of ASR in year of 2009. Is that correct to your understanding.

MR. CARLEY: That is correct, in the Bravo electrical tunnel.

MEMBER REMPE: That's okay. I just wanted to make sure. Thank you.

MR. CARLEY: You're welcome.

MEMBER MARCH-LEUBA: With respect this slide, you said NextEra has implemented the two ASR programs, ASR and building deformation. Is that correct?

MR. CARLEY: That is correct.

MEMBER MARCH-LEUBA: And the moment the licensee's amendment request gets signed, you'll be caring for it. Right now, you're doing it on your own. At the moment this licensee's amendment request related to the ASR methodology, correct?

MR. CARLEY: Yes.

MEMBER MARCH-LEUBA: At this point, you're doing it on your own. At the moment that LRA gets signed, you will be able to take care for it.

MR. CARLEY: We'll be able to close our PODs that are related to building deformation.

MEMBER MARCH-LEUBA: Okay. The real question is on Unit 41, after you get the LRA, anything will change, or everything will be solved before that?

MR. CARLEY: Everything -- we do have a

commitment. And I apologize, I do not remember the commitment number. We have two structures that are non-seismic Category 1 structures, intake and discharge structures, that we have committed to analyze before 2020 and will implement the program for those structures when that analysis is done. So that'll be --

MEMBER MARCH-LEUBA: So the only --

MR. CARLEY: -- a couple of years prior.

MEMBER MARCH-LEUBA: The only change that will happen on Unit 41 will be those addition of two additional non-Category 1 structures?

MR. CARLEY: Those will actually be incorporated in 2020.

MEMBER MARCH-LEUBA: Before the LRA gets issued.

MR. CARLEY: No, before the period of extended operation.

MEMBER MARCH-LEUBA: Correct. That's the appropriate terminology. Okay, thank you.

MEMBER SKILLMAN: Ed, anything else?

MR. CARLEY: With that, I'll turn it over for concluding remarks.

MEMBER SKILLMAN: Okay.

MR. COLLINS: With regards to our

concluding remarks, as presented, Seabrook is committed to the continuous improvement and long term operation of NextEra Seabrook Station. Seabrook will manage the effective agency in accordance with 10 C.F.R. 5421(a)(1). Seabrook has conducted time limited aging analysis that require evaluation under 10 C.F.R. 5421(c).

In summary, in closing, NextEra Energy Seabrook has demonstrated compliance with the requirement of 10 C.F.R. 54 for issuance of a renewed license for Seabrook Station, Unit 1.

This concludes our presentation at this time. I'll turn it over to Ken Browne.

MR. BROWNE: Now, as Mr. Collins noted, Mr. Chairman, that concludes NextEra's presentation for license renewal.

MEMBER SKILLMAN: Okay, Seabrook team, anything else? No? Call-ins, any questions for the Seabrook team before we change out to the NRC team?

(No audible response.)

MEMBER SKILLMAN: Seabrook team, thank you. Please stay in the room. And we call out the NRC team.

(Pause.)

MEMBER SKILLMAN: Thank you, Kendra.

Butch? Take it away, please.

MR. BURTON: All right. Good afternoon. Chairman Corradini, Chairman Skillman, and members of the ACRS. My name is Butch Burton, and I am the license renewal project manager for the Seabrook Station, Unit 1 Safety Review.

We're here today to discuss the staff's review of the Seabrook License Renewal Application which we -- otherwise known as the LRA, as documented in the safety evaluation report that was issued on September 28, 2018.

Joining me here at the table today are Dr. Allen Hiser, senior technical advisor in NRR's Division of Materials and License Renewal, and Mr. Eric Oesterle, branch chief of the projects branch in the division.

Also seated in the audience and available on the phone are members of the NRC technical staff who participated in the review of the license renewal application and conducted onsite audits and inspections.

The presentation is short and sweet. I'll begin the presentation with a general overview of the staff's review. And since there are no open or confirmatory items in the SCR, we'll then proceed to the staff's conclusions.

On May 25th, 2010, NextEra Energy Seabrook submitted an application for renewal of the Seabrook operating license for an additional 20 years or until March 15th, 2050. For the review of the Seabrook license renewal application, the following audits and inspections were conducted onsite.

First, in September 2010, the staff conducted an audit to review NextEra's administrative controls governing the scoping and screening methodology and the technical basis for the scoping and screening results. The staff documented the scoping and screening methodology audit results in a report dated February 4th, 2011.

Second, during two weeks in October 2010, the staff audited NextEra's Aging Management Programs, which we call AMPs, and relayed a documentation to verify NextEra's claim that the programs were consistent with those described in the NRC's Generic Aging Lessons Learned or GALL report and, considering any enhancements or exceptions to the AMPs, whether the programs were adequate to manage aging during the period of extended operation.

The staff considered plant conditions and operating experience during the audits and documented the results in a report dated March 21st, 2011.

Third, during three weeks in March and April 2011, Region I inspectors conducted a 71002 inspection in support of the review of the Seabrook LRA and documented the results in a report dated May 23rd, 2011.

Fourth, during the last week of April 2018, Region I inspectors conducted a second 71002 inspection on Aging Management programs for concrete structures affected by alkali silica reaction, known as ASR. Region I documented the results of this focused inspection in a report dated August 10th, 2018. And this issue was discussed with the ACRS Subcommittee on Plant License Renewal at its October 31st meeting.

In June 2012, the staff issued a safety evaluation report for the Seabrook LRA with seven open items which are listed on this table. In September of 2018, the staff issued a second safety evaluation report which resolved these seven open items.

Following issuance of the SER with open items, the staff and NextEra met with the ACRS Subcommittee on Plant License Renewal in July 2012 to discuss the staff's findings. Of the seven open items documented in the SER, the open item associated with the structure's monitoring program, and how it manages aging associated with ASR, dominated the discussions between the ACRS Subcommittee, NextEra and the staff.

The resolution and closure of the seven open items was documented in the staff's SER issued in September of 2018. During the staff's in depth technical review of the LRA over the last eight years, including two audits and two inspections, a total of 291 RAIs were issued, 58 of which were follow-up RAIs.

Following issuance of the SER in September 2018, the ACRS Subcommittee on Plant License Renewal held meetings with the NRC staff and NextEra, as I mentioned, on October 31st and on November 15th, 2018.

The October 31st meeting was focused on ASR at Seabrook including resolution of the open item associated with the structure's monitoring program, and how the aging effects on structures and components affected by ASR would be managed during the period of extended operation. The November 15th subcommittee meeting focused on the closeout of the remaining open items.

SER Section 2 describes the scoping of systems, structures, and components, known as SSCs, and screening of structures and components to identify those subject to an aging management review, known as an AMR. The staff reviewed NextEra's scoping and screening methodology, procedures, quality controls applicable to the development of the LRA, and training of its project personnel.

The staff also reviewed the various summaries of safety related SSCs, non-safety related SSCs affecting safety functions, and SSCs relied upon to perform functions applicable to Seabrook in compliance with the Commission's regulations for fire protection, environmental qualification, station blackout, and anticipated transients without scram.

Based on its review, results from the scoping and screening audit and additional information provided by NextEra, the staff concludes that NextEra's scoping and screening methodology and its implementation were consistent with the standard review plan for license renewal, known as the SRP, and the requirements of 10 C.F.R. 54.4(a).

SER Chapter 3 and its subsections cover the staff's review of NextEra's programs for managing aging in accordance with 10 C.F.R. 5421(a)(3). Sections 3.1 through 3.6 include the AMR items in each of the general system areas within the scope of license renewal. For a given AMR item, the staff reviewed the item to determine whether it is consistent with the GALL report.

For the AMR items not consistent with the GALL report, the staff reviewed NextEra's evaluation to determine whether NextEra has demonstrated reasonable assurance that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation, as required by 10 C.F.R. 5421(a)(3).

The license renewal application was submitted in 2010 and described a total of 42 Aging Management Programs, 13 of which were new and 29 of which were existing. As a result of the staff's review, two additional plant-specific Aging Management Programs, the ASR Monitoring Program and the Building Deformation Monitoring Program, were developed to address the management of structures affected by ASR, for a total of 44 Aging Management Programs.

All AMPs, with the exception of the plant-specific AMPs, were evaluated by the staff for consistency with Revision 2 of the GALL report. For the plant-specific AMPs, the staff evaluated them against the program elements defined in Appendix A.1 of the SRP.

Section 4 of the SER identifies time

limited aging analyses, or TLAAs. Section 4.1 documents the staff's evaluation of NextEra's identification of applicable TLAAs. The staff evaluated NextEra's basis for identifying those plant-specific or generic analyses that need to be identified as TLAAs and determined that NextEra has provided an accurate list of TLAAs as required by 10 C.F.R. 5421(c)(1).

Section 4.2 through 4.7 document the staff's review of the applicable TLAAs as shown. Based on its review, and the information provided by NextEra, the staff concludes that either the analyses remain valid for the period of extended operation, or the analyses have been projected to the end of the period of extended operation, or the effects of aging on the intended functions will be adequately managed for the period of extended operations as required by 54(c)(1), Subparagraphs I, ii, and iii.

The staff's reviewed NextEra's responses to the open items identified in the safety evaluation report with open items that was issued in June 2012 and finds that all the open items have been satisfactorily resolved and closed. With the closure of the open items, the staff finds that NextEra has met the requirements of 10 C.F.R. 5429(a) for the license renewal of Seabrook Station, Unit 1.

More specifically, the staff finds that actions have been identified and have been or will be taken at Seabrook Station, Unit 1 such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis and that any changes made to the plant's current licensing basis are in accordance with the Atomic Energy Act and the Commission's regulations.

This concludes the staff's presentation, and we'll be happy to take any remaining questions you may have.

MEMBER SKILLMAN: Butch, thank you. Dr. Hiser, Eric, thank you.

Colleagues, any questions for the NRC team, please?

(No audible response.)

MEMBER SKILLMAN: If not, I would ask you to stand by. Let's go to the public. Are there any individuals in the room that would care to make a comment? If so, I invite you to come to the microphone and ---

(Telephonic interference.)

MEMBER SKILLMAN: -- I ask you to come to the microphone, and introduce yourself and speak clearly into the microphone, please.

Seeing none, we go to the phone line. Ladies and gentlemen on the phone line, if one or some of you are out there, would you just please simply say hello so that we know that you are there?

MR. OSSING: Hello?

MEMBER SKILLMAN: Thank you. All right. For any individual on the phone line that would like to make a comment, please introduce yourself and then make your comment, please.

MR. OSSING: Hello, my name is Michael Ossing from Marlborough, Massachusetts. I'd first like to acknowledge the efforts by the NRC staff, and the ACRS, as well as NextEra during this eight-year process.

Seabrook is in compliance with the license renewal and Aging Management Program position --- and positioned, rather, for the station to operate safely during the license renewal process. I would support the ACRS providing a favorable recommendation to issue Seabrook a license renewal for the period of extended operation. Thank you.

MEMBER SKILLMAN: Thank you, sir. Is there another individual out there that would like to make a comment, please?

(No audible response.)

MEMBER SKILLMAN: Hearing none, please close the phone line. And Chairman Corradini, back to you.

CHAIRMAN CORRADINI: Well, thank you. I was expecting there would be more public comments. Okay, thank you very much to NextEra and the staff. And we're going to go off the record, take a couple of minutes to rearrange, and we will probably take up the NextEra letters. And, Dick, you'll lead us through.

MEMBER SKILLMAN: Yes. Let me make one comment. We are going to process two letters this afternoon, we hope. One letter is on the license renewal amendment that is plus 20 years. And the second letter is devoted to Alkali-Silica Reaction. And our desire is to process the ASR letter first and then the license extension letter second. So that's the plan going forward. And we're prepared. Thank you.

CHAIRMAN CORRADINI: We'll take a few minutes to kind of rearrange.

(Whereupon, the above-entitled matter went off the record at 3:51 p.m.)

TVA Clinch River SMR Project Early Site Permit Application

December 6th, 2018

Advisory Committee on Reactor Safeguards

Full Committee Meeting

Acknowledgement and Disclaimer

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Presentation Outline

- Clinch River Nuclear Site – Overview
  - Dan Stout
- Early Site Permit Application – Overview
  - Ray Schiele
- Emergency Preparedness
  - Archie Manoharan

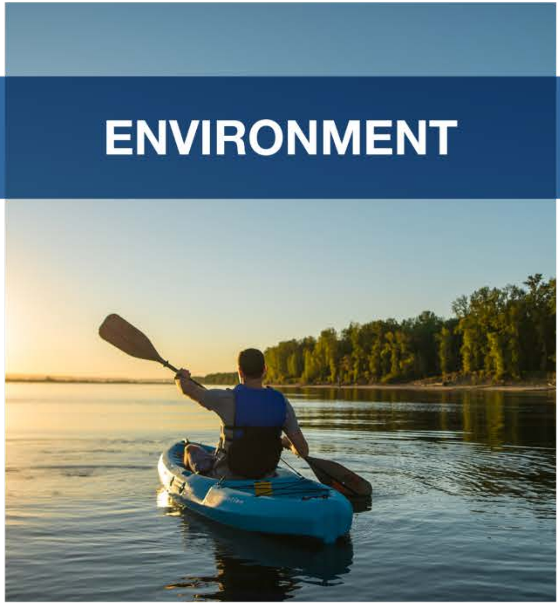
Clinch River Nuclear Site - Overview

Dan Stout

Director, Nuclear Technology & Innovation

TVA's Mission

Provide ***affordable, reliable*** power.



Steward the Valley's ***natural resources***.

Partner for ***economic growth***.

TVA Clinch River Site Site

- Access to 500 KV and 161 KV transmission
- Neighbor to DOE, an interested customer
- Basic Infrastructure
- Abundant and skilled workforce
- Strong community support
- TVA owned/controlled

Early Site Permit Application (ESPA)

An Early Site Permit assesses site suitability for potential construction and operation of a nuclear power plant.

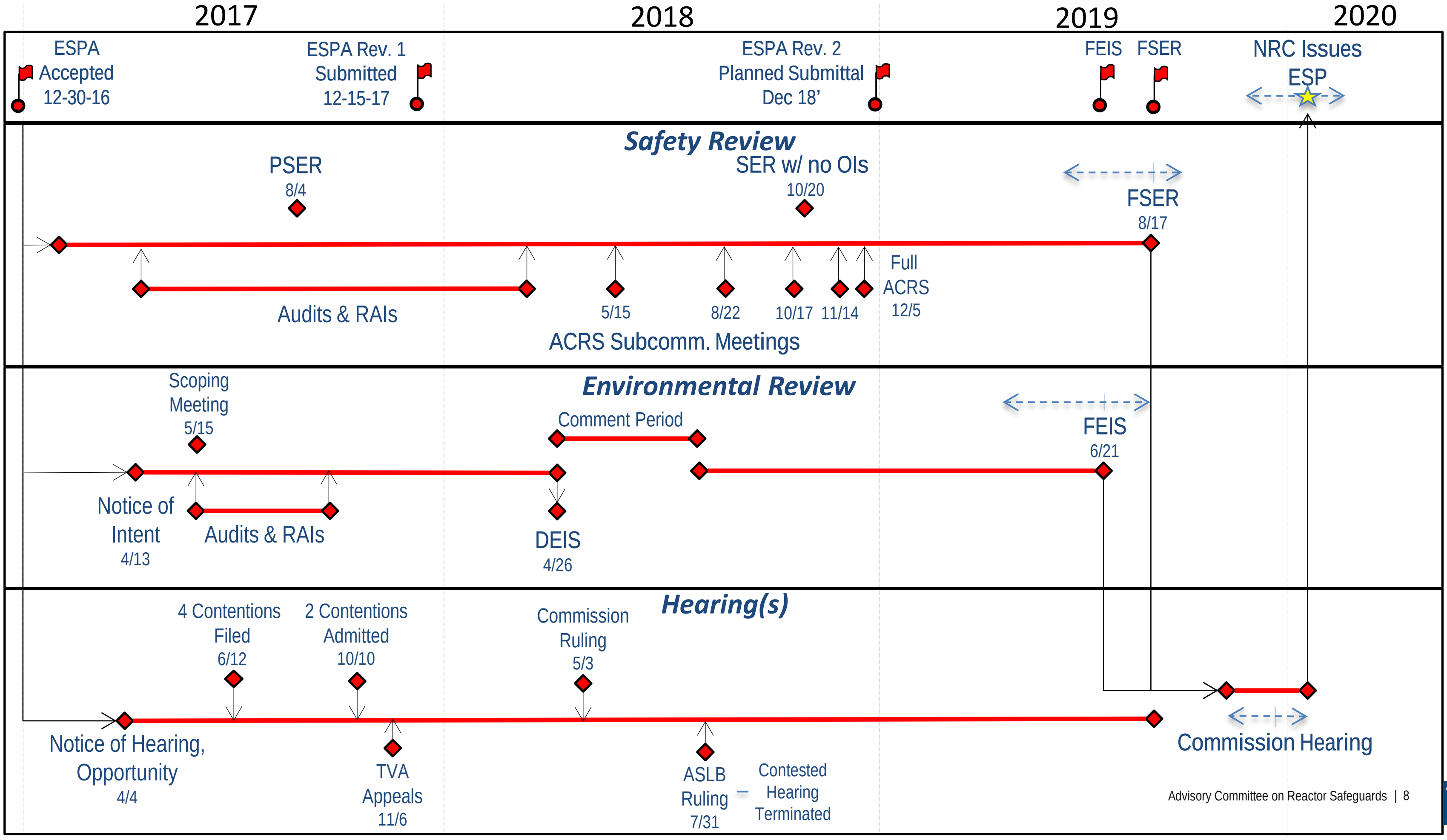
Application includes:

- Site Safety Analysis Report to address impacts of the environment on the plant
- Environmental Report
- Emergency Plans (Part 5A and Part 5B)
- Exemptions (Part 6)

ESPA based on a “plant parameter envelope” (PPE)

- Composite of reactor and engineered parameters from four U.S. light-water SMR designs with unique design features that bound the safety and environmental impact of plant construction and operation
- Developed based on NEI 10-01 guidance with margin added to specific parameters
- Assumes two or more SMR units of a single design
- Up to 800MWt for a single unit with a combined nuclear generating capacity not exceeding 2420 MWt (800 MWe)

NRC Review of ESPA



ESPA Summary

- NRC Commenced Review in FY 17'
- Contains more than 8000 Pages
- Supported by over 80,000 pages in referenced documents
- Efficient Use of Audits
- Few Requests for Additional Information (RAIs)
- Frequent, Clear, and Candid Communication

Early Site Permit- Overview

Ray Schiele

Licensing Manager

Application Organization

Part 1 – Administrative Information

Part 2 – Site Safety Analysis Report

- Chapter 1 – Introduction and General Description
- Chapter 2 – Site Characteristics
- Chapter 3 – Aircraft Hazards
- Chapter 11 – Radioactive Waste Management
- Chapter 13 – Emergency Planning
- Chapter 15 – Transient and Accident Analysis
- Chapter 17 – Quality Assurance

Part 3 – Environmental Report

Part 4 – Limited Work Authorization – Not Used

Part 5 – Emergency Plan

Part 6 – Exemptions and Departures

Part 7 – Withheld Information

Part 8 – Enclosures

ESPA Development

Regulatory bases for the SSAR:

- NRC Regulations—10 CFR 20, 10 CFR 50, 10 CFR 52, and 10 CFR 100
- NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition
- NRC Regulatory Guide 1.206, Combined License Applications for Nuclear Power Plants (LWR Edition)
- RS-002, Processing Applications for Early Site Permits

Regulatory bases for the ER:

- National Environmental Policy Act,
- NRC Regulations—10 CFR 51 and 10 CFR 52,
- NRC Regulatory Guide 4.2, Preparation of Environmental Reports for Nuclear Power Stations,
- NRC Regulatory Guide 4.7, General Site Suitability Criteria for Nuclear Power Stations,
- NUREG-1555, Federal, regional, state and local environmental statutes, as applicable, and
- RS-002, Processing Applications for Early Site Permits.

ESPA NRC Interactions

▪ Pre-Environmental Report Visit	March 2013
▪ PPE Development	September 2014
▪ Pre-application Site Visit	October 2014
▪ Alternative Sites Visit	June 2015
▪ ESPA Readiness Review	August 2015
▪ Hydrology and Health Physics Audit	April 2017
▪ Seismic/Geotechnical Audit	May 2017
▪ Environmental and Meteorology Audit	May 2017
▪ QA Inspection	April 2018
▪ Meteorology and Health Physics Audit	May 2018

ASER/ACRS Committee Timeline

▪ 1st Set ASERs Issued	April 2018
▪ ACRS Subcommittee Meeting	May 2018
– SSAR Sections 2.1, 2.2, 3.5.1.6, 15.0.3	
▪ ASER – SSAR 13.3 Issued	July 2018
▪ ACRS Subcommittee Meeting	August 2018
– SSAR Section 13.3	
▪ ASER – SSAR 2.5 Issued	September 2018
▪ ACRS Subcommittee Meeting	October 2018
– SSAR Section 2.5	
▪ 2nd Set ASERs Issued	October 2018
▪ ACRS Subcommittee Meeting	November 2018
– SSAR Sections 2.3, 2.4, 11.2/11.3, 17.0	
▪ ACRS Full Committee Meeting	December 2018

Emergency Preparedness

Archie Manoharan

Licensing Engineer

ESPA – Emergency Preparedness Approach

Emergency Planning (EP) Information Layout – 3 Areas

- Part 2, SSAR, Section 13.3, Emergency Preparedness
  - Plume exposure pathway (PEP) emergency planning size (EPZ) sizing methodology

- Part 5, Emergency Plan
  - Two major features (Onsite) Emergency Plans
  - Part 5A – Site Boundary EPZ Emergency Plan
  - Part 5B – 2-Mile EPZ Emergency Plan

- Part 6, Exemptions and Departures
  - 2 sets of exemption requests
  - Exemption requests for a PEP EPZ at Site Boundary
  - Exemption requests for a 2-mile PEP EPZ

The final EPZ size for the Clinch River Site will be determined at COLA stage

PEP EPZ Sizing Methodology

- Takes SMR design and safety advancements into consideration
- Dose-based, consequence-oriented approach to determine an appropriate EPZ size
- Consistent with the NUREG-0396 sizing rationale – spectrum of accidents are addressed
- **Approach has the same dose criteria as NUREG-0396 – 1 rem total effective dose equivalent (TEDE)**

Technical Criteria - PEP EPZ should:

- **Criterion A** – encompass those areas in which projected dose from **design basis accidents** (DBAs) could exceed the U.S. Environmental Protection Agency (EPA) early phase protective action guide (PAG)

- **Criterion B** – encompass those areas in which consequences of **less severe core melt accidents** could exceed the EPA early phase PAG

- **Criterion C** – be of sufficient size to provide for substantial reduction in early health effects in the event of **more severe core melt accidents**

PEP EPZ Sizing Methodology

- Step 1 - Accident scenario selection
  - DBA from Chapter 15
  - Design and site specific Probabilistic Risk Assessment (PRA) for severe accident scenarios
    - Considers – all modes, internal & external events, applicable fuel handling, spent fuel pool, and multi-module accidents
    - Sequences with mean core damage frequency (CDF) greater than 1E-8 per reactor-year (rx-yr)
    - Criterion B: Less severe core melt scenarios – Mean CDF greater than 1E-6 per rx-yr, intact containment
    - Criterion C: More severe core melt scenarios – Mean CDF greater than 1E-7 per rx-yr, containment bypass or failure
- Step 2 - Determine source term releases from selected accidents
- Step 3 - Calculate dose consequences at distance
- Step 4 - Compare the dose at distance to EPA early phase PAG

COL applicant would perform an analysis using the PEP EPZ size methodology, with site- and design-specific input, to justify the PEP EPZ size for the COLA

PEP EPZ Sizing Methodology – Example Analysis

Criteria A & B: DBA and less severe accidents

- Dose consequences do not exceed the early phase EPA PAG – 1 rem total effective dose equivalent (TEDE)

Criterion C: More severe accidents

- Calculate distance at which conditional probability to exceed 200 rem whole body exceeds 1E-3 per rx-yr
- Verify the PEP EPZ is of sufficient size to provide for substantial reduction in early health effects

Design-Specific Example Analysis – Evaluates NuScale Power Plant at Clinch River Site

Criteria	Site Boundary Dose TEDE (rem)	EPA Early Phase PAG Limit TEDE (rem)
A: Design Basis Accidents	0.104	1
B: Less Severe Core Melt Accidents	0.158	1
C: Reduction in Early Severe Health Effects	No accident scenarios met the required screening criteria.	

Part 5 – Emergency Plan

Part 5 of the ESPA contains the major features of two distinct Emergency Plans for Clinch River Site in accordance with 10 CFR 52.17(b)(2)(i).

Part 5A

- Describes major features of an Emergency Plan for a PEP EPZ consisting of the area encompassed by the Site Boundary.

Part 5B

- Describes major features of an Emergency Plan for a PEP EPZ consisting of an area approximately two miles in radius surrounding the Clinch River Site.

Both plans address the 16 planning standards in NUREG–0654, Section II, which reflects the requirements in 10 CFR 50.47(b)(1) through 10 CFR 50.47(b)(16) and Appendix E to 10 CFR Part 50 considering the requested exemptions described in Part 6 of the ESPA

Part 6 – Exemptions and Departures

Pursuant to 10 CFR 52.7, Specific Exemptions, which is governed by 10 CFR 50.12, Specific Exemptions, TVA requested exemptions from the following emergency preparedness requirements for the Clinch River Site:

- Certain standards in 10 CFR 50.47(b) regarding onsite and offsite emergency response plans for nuclear power reactor
- Certain requirements of 10 CFR 50.33(g) and 10 CFR 50.47(c)(2) to establish PEP EPZ for nuclear power plants
- Certain requirements of 10 CFR Part 50, Appendix E, which establish the elements that make up the content of emergency plans

Two Sets of Exemptions

- Exemptions for a PEP EPZ established at the **Site Boundary**
    - Deviate from 10-mile PEP EPZ
    - Various elements of a formal offsite emergency plan
    - Evacuation time estimates
    - Certain elements of offsite notifications and exercises
- Exemptions for an approximate **2-mile** PEP EPZ
    - Deviate from 10-mile PEP EPZ

Emergency Preparedness Approach – Summary

	ESPA	COLA
PEP EPZ Methodology (Part 2, SSAR, Section 13.3)	Approval of the dose-based, consequence oriented methodology for determining the PEP EPZ size	Approval of design specific implementation of the methodology approved in the ESPA
EPZ Size (Part 6)	Approval to deviate from the current 10-mile PEP EPZ requirements based on the methodology to determine PEP EPZ size	Approval of design specific PEP EPZ size based on design specific implementation of the methodology
Emergency Plan (Part 5)	Approval of the major features of the Site Boundary and 2-mile emergency plans presented in Part 5	Approval of the remaining elements of either the Site Boundary or 2-mile emergency plans OR a new plan based on design specific PEP EPZ size using methodology

TVA

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Presentation to the ACRS Full Committee Clinch River Nuclear Site - Early Site Permit Application (ESPA) Safety Review

December 6, 2018

Mallecia Sutton, Project Manager, NRO/DLSE/LB3

Allen Fetter, Project Manager, NRO/DLSE/LB3

Section 13.3 Emergency Planning

Michelle Hart, Technical Reviewer, NRO/DLSE/RPAC

Bruce Musico, Technical Reviewer, NSIR/DPR/RLB

Clinch River Nuclear Site ESP Application Review Overview

- Tennessee Valley Authority (TVA) submitted an ESPA for the Clinch River Nuclear Site to NRC (May 26, 2016)
- Application accepted for docketing and detailed technical review on December 30, 2016. Federal Register Notice on acceptance decision (January 12, 2017)
- TVA requested permit approval for a 20-year term along with approval for a plume exposure pathway (PEP) emergency planning zone (EPZ) sizing methodology, 2 major features (onsite) emergency plans, and exemption requests for site boundary and 2-mile PEP EPZs
- Plant Parameter Envelope (PPE) based on four small modular reactor (SMR) designs

Staff Review

- Staff overview presentation to ACRS on ESP, PPE and Clinch River ESP review schedule (November 15, 2017)
- NRC Staff's safety review of the application included 5 audits and 1 inspection, and issuance of 12 request for additional information (RAIs) (comprising 50 questions)
- Staff completed all Advanced Safety Evaluations (ASEs) with no Open Items and presented to ACRS Subcommittee (May 15, 2018 – November 14, 2018)
- ASEs include 42 combine license application (COL) Action Items and 8 Permit Conditions
- Staff cooperated with U.S. Army Corps of Engineers, consulted with Federal Emergency Management Agency, and engaged with U.S. Department of Energy, Tennessee Department of Environment and Conservation, the U.S. Geological Survey and the Tennessee Emergency Management Agency

ESP Plant Parameter Envelope

Approving an ESP Site without a Selected Reactor Technology

- ESP Plant Parameter Envelope (PPE) values can bound a variety of reactor technologies rather than one specific technology (an amalgam of values representing a surrogate nuclear plant)
- The PPE values are bounding criteria used by staff to determine the suitability of an ESP site for construction and operation of a nuclear plant
- In the combined license application (COLA), when a specific technology is identified, the PPE values are compared to those of the selected technology. If design parameters of the selected technology exceed bounding ESP PPE values, additional reviews are conducted to ensure that the site remains suitable from a safety and environmental standpoint for construction and operation of the selected nuclear plant technology

ESP Plant Parameter Envelope (cont'd)

TVA used the following reactor designs to develop the Plant Parameter Envelope (PPE):

- BWXT mPower SMR, 530 megawatts thermal (MWt) (180 megawatts electric (MWe)
- NuScale SMR, 160 MWt (50 MWe)
- Holtec SMR-160, 525 MWt (160 MWe)
- Westinghouse SMR, 800 MWt (225 MWe)

TVA's PPE is based on construction and operation of two or more SMRs at the Clinch River Nuclear Site with a maximum site nuclear generating capacity of 2420 MWt (800 MWe)

Safety Evaluation Sections

Chapter Sections	Accession Numbers
2.1 Geography and Demography	ML18102B203
2.2 Nearby Industrial Transportation and Military Facilities	ML18102B203
2.3 Meteorology	ML17289B148
2.4 Hydrologic Engineering	ML17289B151 (NP) ML18290A685 (P)
2.5.1 Geologic Characterization	ML17289B252
2.5.2 Vibratory Ground Motion	ML17289B253
2.5.3 Surface Deformation	ML17289B254
2.5.4 Stability of Subsurface Materials and Foundations	ML17289B255
2.5.5 Stability of Slopes	ML17289B255
3.5.1.6 Aircraft Hazards	ML18102B150
11.2 & 11.3 Radioactive Waste Management	ML17289A625
13.3 Emergency Planning	ML17291A052
15.0.3 Radiological Consequences of Design Basis Accidents	ML18102B149
17.5 Quality Assurance Program Description	ML17291A547

Section 2.1 Geography and Demography

- TVA provided adequate information pertaining to;
  - the site setting and boundaries
  - Exclusion Area Boundary (EAB) authority and control
  - current and future population projections
  - low population zone (LPZ) distance, population center distance and population density
- Based on the information provided by the applicant and staff's independent confirmatory evaluation, the staff found the information to be acceptable as it meets the requirements of 10 CFR 100.20

Section 2.2 Nearby Industrial, Transportation, and Military Facilities

- TVA adequately identified potential sources and hazards in site vicinity
- TVA adequately evaluated potential accidents pertaining to explosions, vapor cloud explosions, hazardous/toxic chemical vapors, and fires
- Based on the information provided by the applicant and staff’s independent confirmatory evaluation, the staff found the information to be acceptable as the information meets the guidance provided in NUREG-0800 Section 2.2.1-2.2.2

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Section 2.3 - Meteorology

- Site characteristics related to **extreme weather** (hurricane and tornado winds, winter precipitation, temperature and humidity extremes) are acceptable

- Onsite **meteorological monitoring system** provides adequate data to represent meteorological dispersion conditions

- Site characteristics related to **Short-Term (Accident)** and **Long-Term (Routine Release) dispersion estimates** (X/Q and D/Q values) are acceptable

- Based on the information provided by the applicant, the staff found all regulatory requirements have been satisfied with no open items

Short-Term (Accident) X/Q Values

- Short-Term (Accident) X/Q Values
  - Exclusion Area Boundary (335 meters)
  - Low Population Zone (1609 meters)
- Based on PAVAN Atmospheric Dispersion Model
  - Gaussian model
  - Various time averaging periods
    - 0-2 hr @ EAB
    - 0-8 hr, 8-24 hr, 1-4 days, and 4-30 days @ LPZ
  - Intended to represent dispersion conditions that are exceeded no more than 5% of the time
- Used to demonstrate compliance with 10 CFR 52.17(a)(1)(ix) dose guidelines for design basis accidents
  - 25 rem at the EAB for any 2-hour period following the onset of the release
  - 25 rem at the outer boundary of the LPZ for the duration of the release

Section 2.4 Hydrologic Engineering

- TVA proposed adequate site characteristics and bounding design parameters for inclusion in the ESP
- Design basis flood and maximum groundwater levels, and the accidental release dose estimate meet regulatory requirements
- Staff concludes that applicant meets ESP regulatory requirements associated with hydrologic engineering

Section 2.5 Geology, Seismology and Geotechnical Engineering

- Geologic Site Characterization (Section 2.5.1) - No tectonic features with the potential for adversely affecting suitability of the site occur in the site region, site vicinity, site area, or at the site location
- Vibratory Ground Motion (Section 2.5.2) - Applicant's ground motion response spectrum adequately represents the regional and local seismic hazards, and accurately includes the potential effects of local site-specific subsurface properties
- Surface Deformation (Section 2.5.3) - Negligible potential exists for tectonic surface deformation at the site. Karst is the primary potential hazard for non-tectonic surface deformation that could adversely affect the site
- Stability of Subsurface Materials and Foundations (Section 2.5.4) - Applicant adequately determined the engineering properties of subsurface materials at the site, and properly evaluated the stability of subsurface materials and foundations based on results of field and laboratory tests and state-of-the-art methodology
- Stability of Slopes (Section 2.5.5) - Applicant provided necessary information on site topography and geologic conditions, and adequately described characteristics of slopes at the site

Section 3.5.1.6 Aircraft Hazards

- For site suitability, aircraft accidents should not lead to radiological consequences in excess of the exposure guidelines of 10 CFR 50.34(a)(1) with a probability of occurrence greater than about 10^{-7} per year
- The applicant determined an aircraft crash probability of 7.53×10^{-7} per year from two nearby airways not associated with local airport operations
- The staff conservatively estimates a potential aircraft crash probability of 1.5×10^{-8} per year (bounding the applicant's probability), assuming all flights within 10 miles of the site follow the two airways passing near the site
- Staff finds that the applicant's approach is reasonable and the probability value is acceptable

Chapter 11 Radioactive Waste Management, Sections 11.2.3 and 11.3.3

- Applicant's methodology to develop the normal PPE liquid and gaseous effluent release source terms for use in calculating offsite doses is reasonable
- Normal PPE liquid and gaseous effluent release concentrations meet the unity rule in 10 CFR Part 20, Appendix B, Table 2, Columns 1 and 2
- Offsite doses from normal PPE liquid and gaseous effluent release source terms meet the design objectives in 10 CFR Part 50, Appendix I, Sections II.A, II.B, and II.C; Environmental Protection Agency's (EPA) radiation standards in 40 CFR Part 190, as implemented under 10 CFR 20.1301(e); and public dose limit in 10 CFR 20.1301
- Reactor designs falling within the normal PPE effluent release source terms and offsite doses for the Clinch River Nuclear Site are without undue risk to public health and safety

Chapter 15 Accident Analysis

- Evaluation of the radiological consequences of postulated design basis accidents (DBAs) is based on the PPE accident source term for DBA isotopic releases to the environment (in lieu of specific plant design information) in conjunction with site characteristic short term (accident) atmospheric dispersion factors

- The same dose criteria are used for siting and postulated accident dose analysis requirements:

  The evaluation must determine that:

  1. An individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the postulated fission product release would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE).
  2. An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem TEDE

- Staff concluded that the applicant's analysis meets the dose criteria specified, and the PPE includes the bounding accident releases for the determination

Section 17.5 Quality Assurance Program Description

- NRC Staff identified one RAI, March 9, 2018
- NRC Staff conducted Quality Assurance Implementation Inspection, April 16-20th 2018.
- TVA issued Nuclear Quality Assurance Plan, Revision 36; May 8, 2018
- Staff concluded that the applicant’s quality assurance program description for the Clinch River Nuclear site ESP application meets the requirements of 10 CFR Part 50, Appendix B and 10 CFR 52.17(a)(1)(xi) and (xii)

13.3 Emergency Planning

The ESPA requested review of 3 key areas, which consist of:

- Plume exposure pathway (PEP) emergency planning zone (EPZ) sizing methodology
- 2 major features (onsite) emergency plans (ESPA Part 5)
  - ESPA Part 5A reflects a site boundary PEP EPZ
  - ESPA Part 5B reflects a 2-Mile PEP EPZ (including an ETE)
- 25 Exemption Requests (ESPA Part 6)
  - 2 exemption requests (applicable to both the site boundary and 2-mile PEP EPZs)
  - 23 exemption requests address portions of 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50 for offsite emergency planning (EP) related to the site boundary PEP EPZ only

25 Exemption Requests (EP)

- 10 CFR 50.33(g) & 50.47(c)
  - 2 requests for exemptions from the 10-mile PEP EPZ requirement
- 10 CFR 50.47 & Appendix E to 10 CFR Part 50
  - 23 requests for exemption from the emergency planning requirements associated with offsite emergency planning
    - State & local emergency plans
    - Public alert & notification
    - Evacuation Time Estimate (ETE)
    - Offsite exercises

Basis for Acceptance

- The ESPA provides a basis for the establishment (in the COLA) of either a Site Boundary or 2-mi PEP EPZ, which maintains the same level of protection (i.e., dose savings in the event of a radiological emergency) in the environs of the Clinch River Nuclear Site as that which exists in the basis for a 10-mi PEP EPZ

Combined License Application

- Upon issuance of the ESP, the applicant acquires approval, with conditions, of:
  - The PEP EPZ sizing methodology
  - The 2 major features emergency plans (site boundary/2-mile PEP EPZ)
  - The 25 requested exemptions
- A COLA that incorporates by reference the ESP must:
  - Identify the chosen SMR technology for the Clinch River Nuclear site
    - Demonstrate that the EPZ sizing methodology supports either the site boundary or 2-mile PEP EPZ
  - Provide a complete & integrated emergency plan
    - 2-mile PEP EPZ must provide onsite & offsite emergency plans
    - site boundary PEP EPZ must provide an onsite emergency plan
  - Address all 16 COL Action Items and 4 Permit Conditions

EPZ Size Determination in COLA

- COL Action Item 13.3-1 (reflects ESPA Part 2 Section 13.3.3.1.4)
  - Identify chosen SMR technology & major features emergency plan
  - Provide detailed information that shows the ability of the SMR to meet the chosen PEP EPZ
  - The selected SMR technology must meet the EPA early phase protective action guide (PAG)
- Permit Condition 1
  - Provide detailed information to demonstrate that the accident release source term information for the PEP EPZ size determination analysis using the selected SMR design is bounded by the non-design-specific plant parameter source term information used in the analysis supporting the exemption requests (ASER Table 13.3-1)
  - Based on non-design-specific bounding 4-day accident release source term that meets EPZ size criteria

TVA PEP EPZ Size Methodology Technical Criteria

- PEP EPZ should encompass those areas in which projected dose from DBAs could exceed the EPA early phase PAG

- PEP EPZ should encompass those areas in which consequences of less severe core melt accidents could exceed the EPA early phase PAG

- PEP EPZ should be of sufficient size to provide for substantial reduction in early health effects in the event of more severe core melt accidents

TVA PEP EPZ Size Methodology SSAR Section 13.3.3.1

- Accident scenario selection
  - Use bounding DBA from COLA Final Safety Analysis Report Chapter 15
  - Use COLA site- and design-specific probabilistic risk assessment to categorize severe accident scenarios
    - All modes, internal and external events, applicable fuel handling and spent fuel pool accidents, multi-module considerations
    - Assess all sequences with mean core damage frequency (CDF) > 10^{-8} per rx-yr
    - More probable, less severe core melt scenarios
      - Mean CDF > 10^{-6} per rx-yr
      - Intact containment
    - Less probable, more severe core melt scenarios
      - Mean CDF > 10^{-7} per rx-yr
      - Includes containment bypass or failure
- Determine source term releases to atmosphere
- Calculate dose consequences at distance from plant
- Determine PEP EPZ size that meets the dose-based criteria

TVA Dose-Based PEP EPZ Size Criteria

- Dose to individual from exposure to the airborne plume during its passage and to groundshine, using average atmospheric dispersion characteristics for site
  - Staff expects the applicant may use the calculation tools used for severe accident consequence analysis in environmental report
- DBA and more probable, less severe accidents
  - 1 rem TEDE from 96-hr exposure
  - Lower end of dose range EPA PAG for early phase protective actions (e.g., evacuation and sheltering)
  - Verify that dose consequences do not exceed the EPA PAG beyond the site boundary (within owner controlled area) and 2-mile PEP EPZs
- Less probable, more severe accidents
  - Calculate the distance at which the conditional probability to exceed 200 rem whole body from 24-hr exposure exceeds 10^{-3} per rx-yr
  - Acute dose at which radiation-induced early health effects may begin to be noted (e.g., nausea)
  - Verify that the PEP EPZ supports substantial reduction in early health effects

Review of PEP EPZ Size Methodology

- Staff compared TVA's methodology and dose criteria to the study used as technical basis for current 10-mile PEP EPZ requirement (NUREG-0396)
  - The features of TVA's methodology are consistent with NUREG-0396
    - Considered a range of accidents
    - Performed accident consequence analyses
    - Determined an area outside of which early protective actions are not likely to be necessary to protect the public from radiological releases
- The staff concludes that the applicant's proposed methodology is reasonable, and consistent with the analyses that form the technical basis for the current regulatory requirement of a PEP EPZ of about 10 miles in radius

EP Exemption Plant Parameters

- TVA developed a non-design-specific accident release source term that would meet the PEP EPZ size criteria to be used as plant parameters (ASER Table 13.3-1)
  - Isotopic total release activity over 96 hrs results in TEDE of about 0.9 rem at site boundary
  - Same idea as PPE DBA source term to envelope an unknown design
  - Referenced in Permit Condition 1 for adoption of EP exemptions

Section 13.3 EP Conclusions

- The staff concludes that:
  - The PEP EPZ sizing methodology is acceptable for determining the appropriate size of the PEP EPZ for the Clinch River Nuclear site because it is consistent with the analyses that form the technical basis for the current 10-mile PEP EPZ
  - The 2 major features emergency plans are acceptable because they meet the applicable standards of 10 CFR 50.47 and requirements of Appendix E to 10 CFR Part 50
  - The exemption requests are acceptable because they are authorized by law, will not present an undue risk to the public health and safety, are consistent with the common defense and security, and special circumstances are present

Questions?

Technical Reviewers

Dan Barss
Luissette Candelario
Yuan Cheng
Richard Clement
Joseph Giacinto
Michelle Hart
David Heeszel
Michael Mazaika
Bruce Musico
Kevin Quinlan
Nicholas Savwoir
Gerry Stirewalt
Seshagiri (Rao) Tammara
Jenise Thompson
Weijun Wang
Jason White

Acronyms

ASE – Advanced Safety Evaluation
CFR – Code of Federal Regulations
COL – Combined License
COLA – Combined License Application
CDF – Core Damage Frequency
CP – Construction Permit
CRN – Clinch River Nuclear
DBA – Design Basis Accidents
DBF – Design Basis Flood
EAB – Exclusion Area Boundary
EP – Emergency Planning
EPA – Environmental Protection Agency
EPZ – Emergency Planning Zone
ESP – Early Site Permit
ESPA – Early Site Permit Application
ETE – Evacuation Time Estimate
FRN – Federal Register Notice
LOCA – Loss of Coolant Accident
LPZ – Low Population Zone
NP-Non-Public
MWe – Megawatts Electric
MWt – Megawatts Thermal
NP-Non-Public

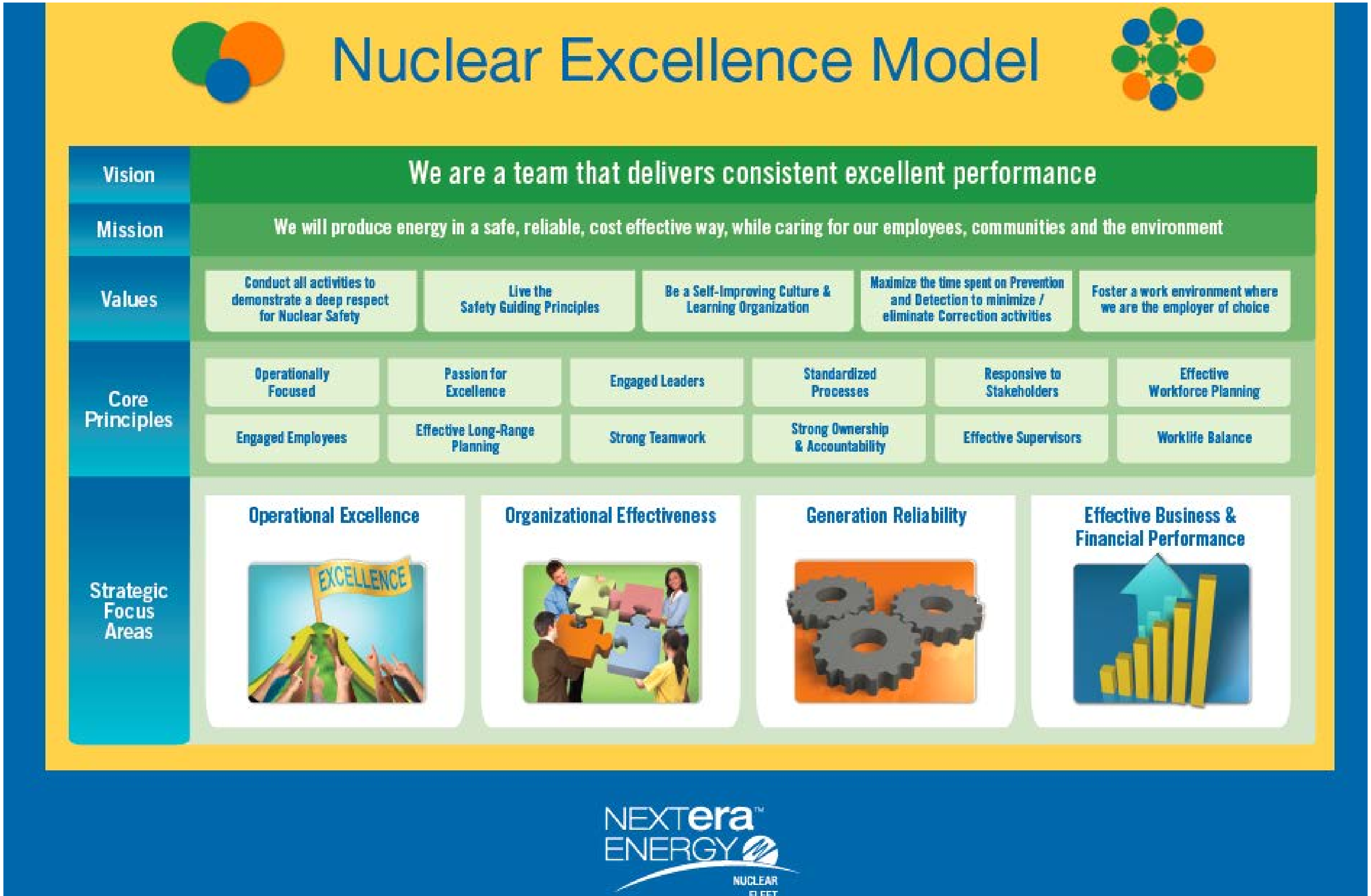
NRC – Nuclear Regulatory Commission
P-Public
PAG – Protective Action Guide
PEP – Plume Exposure Pathway
PPE – Plant Parameter Envelope
RAI – Request for Additional Information
SER – Safety Evaluation Report
SMR – Small Modular Reactor
SSCs – Structures, Systems and Components
TEDE – Total Effective Dose Equivalent
TVA – Tennessee Valley Authority
USGS - U.S. Geological Survey

Seabrook Station Unit 1
License Renewal Application

Advisory Committee on Reactor Safeguards

Full Committee Meeting

December 5, 2018



The foundation for everything we do are the Values and Core Principles of our Nuclear Excellence Model

NEXTera ENERGY
SEABROOK

Agenda

- Introduction
- Site and Station Description
- License Renewal Application and Aging Management Programs
- Safety Evaluation Report and Closure of Previous Open Items
- Closing Remarks

NextEra Energy Seabrook has met the requirements of 10 CFR 54 for issuance of a renewed licensed for Seabrook Station Unit 1

Personnel in Attendance

• Eric McCartney	Regional Vice President - Northern Region
• Michael Collins	Engineering Director
• Ken Browne	Licensing Manager
• Edward Carley	License Renewal Supervisor
• Rudy Gil	Programs Engineering Manager

NEXTera ENERGY SEABROOK

Site and Station Description

Plant Status

- **Completed latest refuel outage (OR19) 10/29/18**
- **Next Refuel Outage – Spring 2020 (End of Cycle 20)**
- **Capacity Factor 15 of 19 cycles > 94%**
  - Lifetime 87%
  - Lifetime excluding refueling outages 95.2%
  - Cycle 19: 99.86%
  - Cycle 18: 98.34%
  - Cycle 17: 99.27%
  - Cycle 16: 99.71%

Capacity factor performance is representative of solid equipment reliability and material condition

NEXTera ENERGY SEABROOK

Recent Station Improvements

- Main Generator Stator Rewind
- Vital Batteries
- Vital Inverters
- Generator Step-Up Transformers
- Mechanical Stress Improvement Process completed for all Reactor Vessel Nozzles
- Service Water Piping (AL6XN)
- Incore Detectors
- Process Control Single Point Vulnerability Circuit Cards
- Solid State Protection System Circuit Cards
- Rod Control Motor/Generator Sets
- Shutdown Reactor Coolant Pump Seals

NextEra Energy Seabrook is committed to maintaining high levels of safety, reliability and performance

License Renewal Application

Scoping and Screening

- Aging Management Review

Time Limited Aging Analysis (TLAA)

UFSAR Supplement

- Commitments

Aging Management Programs

Environmental Report

- Severe Accident Mitigation Alternatives (SAMA) Analysis

NEXTera ENERGY SEABROOK

GALL Consistency

AMPS		Consistent	Consistent with Enhancements	Consistent with Exceptions	Consistent with Exception and Enhancements	Plant Specific
New	15	7	1	2	1	4
Existing	29	8	12	2	5	2
Total	44					

Safety Evaluation Report

SER Issued September 28, 2018

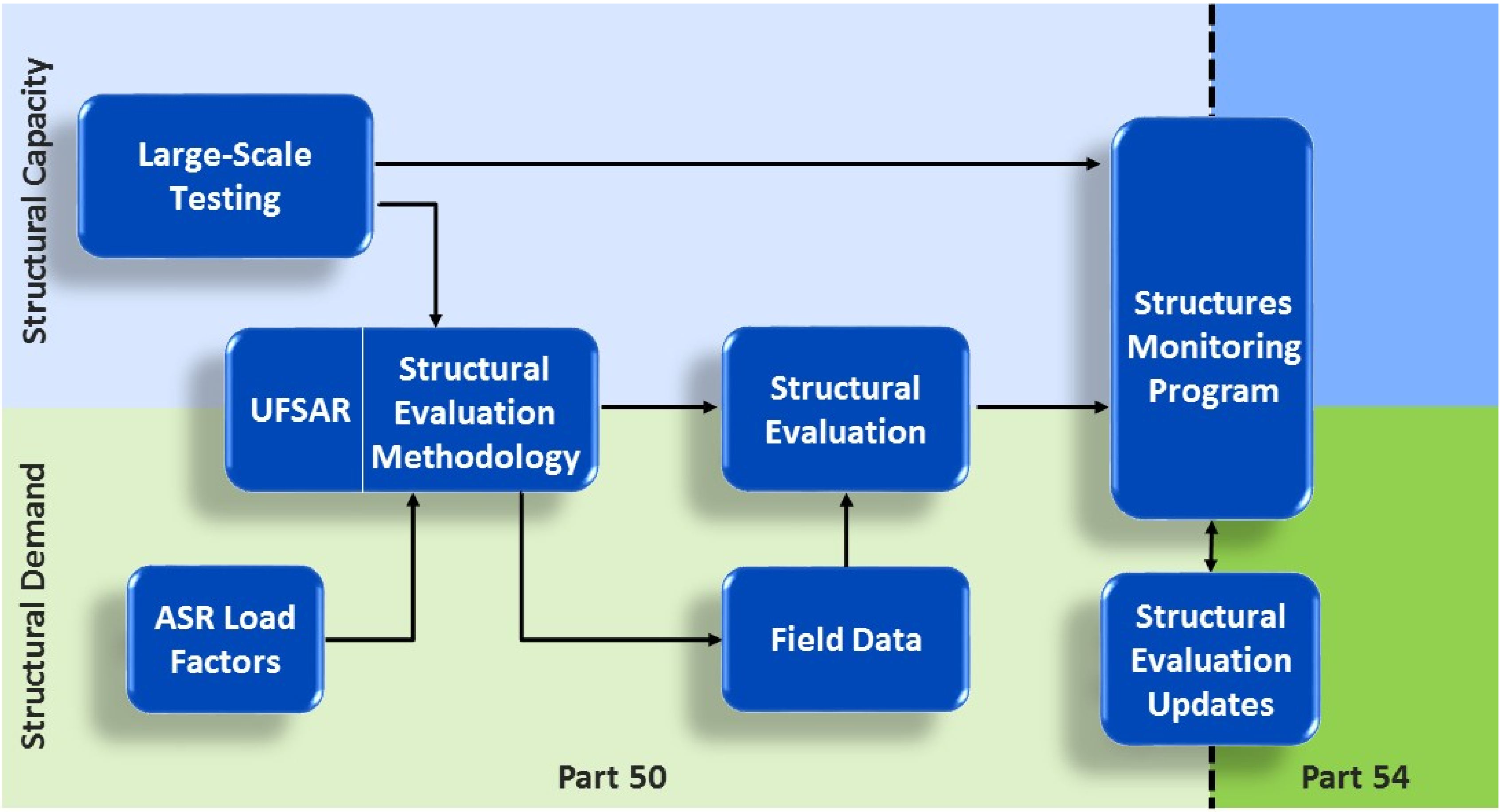
- No open items
- No confirmatory items

Closure of Open Items from previous SER (2012)

- OI 3.0.3.2.2-1— Steam Generator Tube Integrity
- OI 4.2.4-1— Pressure-Temperature Limit
- OI 3.2.2.1-1— Treated Borated Water
- OI 3.0.3.1.7-1— Bolting Integrity Program
- OI B.1.4-2— Operating Experience
- OI 3.0.3.1.9-1— ASME Section XI, IWE Program
- OI 3.0.3.2.18-1— Structures Monitoring Program

NEXTera ENERGY SEABROOK

Approach for Addressing ASR at Seabrook Station



NextEra Energy Seabrook has implemented an effective program for evaluating and managing the impacts of ASR on affected concrete structures and associated SSCs

Concluding Remarks

- Seabrook is committed to the continuous improvement and long-term operation of Seabrook Station
- Seabrook will manage the effects of aging in accordance with 10 CFR 54.21(a)(1)
- Seabrook has evaluated time-limited aging analyses that require evaluation under 10 CFR 54.21(c)
- Seabrook has met the provisions of 10 CFR 54 for issuance of a renewed license

NextEra Energy Seabrook has demonstrated compliance with the requirements of 10 CFR 54 for issuance of a renewed licensed for Seabrook Station Unit 1

NEXTera ENERGY SEABROOK

Advisory Committee on Reactor Safeguards Full Committee

Seabrook Station, Unit 1

Safety Evaluation Report (SER)

December 6, 2018

William “Butch” Burton, Project Manager

Office of Nuclear Reactor Regulation

Presentation Outline

- **Overview of Seabrook license renewal review**
- **Conclusion**

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License Renewal Review: Audits and Inspections Onsite

Audit / Inspection	Dates
Scoping & Screening Methodology Audit (ML110270026)	September 20 – 23, 2010
Aging Management Program (AMP) Audits (ML110280424)	October 12 – 15, 2010 October 18 – 22, 2010
Region I 71002 Inspection: Scoping, Screening, and AMPs (ML111360432)	March 7 – 11, 2011 March 21 – 25, 2011 April 4 – 8, 2011
Region I 71002 Inspection: AMPs for Alkali-Silica Reaction (ASR) (ML18222A292)	April 30 – May 3, 2018

SER Overview

- SER with 7 Open Items issued June 2012
    1. Bolting Integrity Program
    2. ASME Code Section XI, Subsection IWE Program
    3. Steam Generator Tube Integrity Program
    4. Operating Experience
    5. Treated Borated Water
    6. Pressure-Temperature Limit
    7. Structures Monitoring Program/ASR
- Open items closed on September 28, 2018

SER Overview

- SER with 7 Open Items issued June 8, 2012
- Staff met with ACRS Subcommittee on Plant License Renewal on July 10, 2012
- Final SER issued September 28, 2018
  - No open items or confirmatory items
  - Total of 291 RAIs issued
    - 58 follow-up RAIs
- Additional meetings with ACRS Subcommittee on Plant License Renewal held October 31 and November 15, 2018

SER Section 2

- **Structures and Components Subject to Aging Management Review (AMR)**
  - Section 2.1: Scoping and Screening Methodology
  - Section 2.2: Plant-Level Scoping Results
  - Sections 2.3, 2.4, 2.5: Scoping and Screening Results

SER Section 3

- **Aging Management Review (AMR) Results**
  - Section 3.1: Aging Management of Reactor Vessel, Internals, and Reactor Coolant System
  - Section 3.2: Aging Management of Engineered Safety Features
  - Section 3.3: Aging Management of Auxiliary Systems
  - Section 3.4: Aging Management of Steam and Power Conversion Systems
  - Section 3.5: Aging Management of Containments, Structures and Component Supports
  - Section 3.6: Aging Management of Electrical Commodity Group

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SER Section 3

Section 3.0.3 - Aging Management Programs (AMPs)

NextEra's Disposition of AMPs	Final Disposition of AMPs in SER
• 13 new programs – 6 consistent – 1 consistent with enhancements – 3 consistent with exceptions – 3 consistent with enhancements and exceptions	• 15 new programs – 7 consistent – 1 consistent with enhancement – 2 consistent with exceptions – 1 consistent with enhancements and exceptions – 4 plant specific
• 29 existing programs – 10 consistent – 10 consistent with enhancements – 3 consistent with exceptions – 4 consistent with enhancements and exceptions – 2 plant specific	• 29 existing programs – 8 consistent – 12 consistent with enhancements – 2 consistent with exceptions – 5 consistent with enhancements and exceptions – 2 plant specific

SER Section 4

- **Time-Limited Aging Analyses (TLAAs)**
  - 4.1: Identification of TLAAs
  - 4.2: Reactor Vessel Neutron Embrittlement Analyses
  - 4.3: Metal Fatigue Analyses
  - 4.4: Environmental Qualification of Electric Equipment
  - 4.5: Concrete Containment Tendon Prestress Analyses
  - 4.6: Containment Liner Plate, Metal Containment, and Penetrations Fatigue Analyses
  - 4.7: Other Plant-Specific TLAAs

Conclusion

On the basis of its review, the staff finds that the requirements of 10 CFR 54.29(a) have been met for the license renewal of Seabrook Station, Unit 1.