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NUCLEAR REGULATORY COMMISSION

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558TH MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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THURSDAY,

DECEMBER 4, 2008

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ROCKVILLE, MARYLAND

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The Advisory Committee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B3, 11545 Rockville Pike, at 8:30 a.m., Dr. William Shack, Chairman, presiding.

COMMITTEE MEMBERS PRESENT:

WILLIAM J. SHACK, Chair
MARIO V. BONACA, Vice Chair
SAID ABDEL-KHALIK, Member at Large
JOHN SIEBER
SANJOY BANERJEE
JOHN W. STETKAR
J. SAM ARMIJO
DANA A. POWERS

COMMITTEE MEMBERS PRESENT (CONT.)

DENNIS C. BLEY
MICHAEL T. RYAN
OTTO L. MAYNARD
CHARLES H. BROWN, JR.
HAROLD B. RAY
MICHAEL CORRADINI
GEORGE E. ASPOTOLAKIS

ALSO PRESENT:

RICH MILLER
IRA POPPEL
STEVE KIMURA
AMY CUBBAGE
IAN JUNG
DENNIS GALVIN
HULBERT LI
DEREK WIDMAYER
JIM DAVIS
CHRISTIAN ARAGUAS
HOSUNG AHN
GARY STIREWALT
SARAH GONZALES
CLIFF MUNSON

BRUCE MUSICO

ALSO PRESENT: (CONT.)

WEIJUN WANG
BRET TEGELER
CARL CONSTANTINO
JOHN MA
PAUL CLIFFORD
KEN YUEN
WILLIAM RULAND
RALPH MEYER
RICHARD LOBEL
MARTY STUTZKE
BOB DENNIG
JACK ROWE
ZEYNA ABUDLLAHI

DESIGNATED FEDERAL OFFICIAL:

SAM DURAISWAMY

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PROCEEDINGS

CHAIR SHACK: The meeting will now come to order. This is the first day of the 558th meeting of the Advisory Committee on Reactor Safeguards.

During today's meeting the committee will consider the following:

Chapters 7 and 14 of the SER associated with the ESBWR design certification application; early site permit application and the final SER for the Vogtle Nuclear Plant; status of staff activities associated with potential revision to 10 CFR 50.46(b); and the NRC staff's initial white paper on containment overpressure credit issue; and preparation of ACRS reports.

A portion of the session dealing with the ESBWR design certification application may be closed to protect proprietary information applicable to this matter.

This meeting is being conducted in accordance with the provisions of the Federal Advisory Committee Act.

Mr. Sam Duraiswamy is the Designated Federal Official for the initial portion of the meeting.

We have received no written comments or

requests for time to make oral statements from members of the public regarding today's session.

Representatives of the Tennessee Valley Authority will be on the phone bridge line to listen to the discussion regarding the staff's initial white paper on containment overpressure credit.

To preclude interruption of the meeting, the phone line will be placed in a listen-in mode during the presentations and committee discussion.

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.

Our first item this morning is chapters 7 and 14 of the SER, design certification, and Mike will be leading us through that.

MEMBER CORRADINI: Thank you, Mr. Chairman.

So for all the members, just to remind you where we are, we had a subcommittee meeting yesterday afternoon on chapter 7. This was kind of a postponed subcommittee meeting. We had originally scheduled both 14 and 7 in October, and because of some procedural things we weren't able to cover chapter 7 at that time.

So yesterday we went over chapter 7 with open items, of the SER, that is, and GEH and the staff presented where they are on those issues. So we're going to have essentially a more summarized version of that today. I didn't want to take up six hours of your day when we're short of time.

So we'll start with Mr. Ira Poppel from GEH will lead the discussion and Rich Miller and Steve Kimura are both here to join in as necessary.

Ira.

MR. POPPEL: My name is Ira Poppel. I work for GEH, and I am involved in the CNI Group and the overall configuration of the ESBWR DCIS.

Previously I had done similar things for the Lungman project on the ABWR.

This is a necessarily abbreviated presentation, but we just want to give you an overview of what the DCIS looks like.

MEMBER APOSTOLAKIS: Would you please spell out the acronyms the first time you use them?

MR. POPPEL: Oh, I --

MEMBER APOSTOLAKIS: I think there is a number of them coming up. So DCIS, what is that?

MR. POPPEL: Distributed control and

information systems.

MEMBER APOSTOLAKIS: Thank you very much.

MR. POPPEL: The ESBWR has very few of the traditional hard wired switches, panel meters, recorders, indicators, and controllers. It's essentially in both the safety and nonsafety side, for want of a better word, computer type controllers.

This is an overview of the DCIS. There are several ways to present what the system looks like. This is a very, very broad functional overview. The interconnections are functional; they are not meant to be specific.

In the lower left side you can see the four safety divisions -- QDCI -- we call that QDCIS, safety DCIS, and they are organized such that we have multiplexing coordinate in the field, the field being just the reactor building for the safety equipment. And in the control building in what is sometimes referred to as the back panel area or the QDCIS rooms, we have four divisional QDCIS rooms that are in separate fire zones from each other and from the main control room.

The safety-related equipment is -- this is another way of looking at the same picture, and in this case we have -- and still in the lower left is

QDCIS, but what this chart adds to it is what functions are being accomplished by what boxes. Again, functionally.

So if you look at QDCIS, you can see that we are doing the reactor trip, we are doing neutron monitoring, we are doing ECCS systems and safety-related information systems.

It connects through gateways. Basically there is physical, electrical, and data isolation of the signals going from safety to nonsafety, and then it goes into the nonsafety system.

The nonsafety systems are organized into segments, so the important concept is in a traditional DCIS you can refer to concepts like "the network."

In fact, there are five networks, each of which are dual redundant in this system, and although they are not the same sort of isolation you would think about with safety, the networks -- the five individual networks are capable of working independently of one another and do not need information from any of the other networks.

So, for example, in the unlikely event the dual redundant PIP -- it means plant investment protection -- network goes down, PIP A, it does not affect the operation, control, monitoring of the segment equipment and information on PIP B.

So normally if you look at the main control room, all of these nonsafety displays here are available to do any nonsafety function that the operator wants. But as parts of the networks degrade, as these things fail, it will gracefully fail into so some displays won't work but others will.

So normally the segmentation is transparent to the operator, but we are very well assured of the fact that the system will degrade very gracefully.

There is a balance of plant network which is what you would consider to be traditional power generation type stuff, okay, turbine generator control, et cetera.

All of the equipment on these networks is dual redundant at least; some are triply redundant. So we expect a lot less transients to be caused by the C&I system because there are no single failures in the C&I portion, control instrumentation portion of the design. Okay.

And this is generally called the unit data highway, but it's basically the collection of the five network segments, and we have a bridge to the plant data highway which is basically lesser important things, like printers, okay.

So you might consider this the control network, and you might consider this like a utility type network.

The control network is unique also in that the data on it is not ethernet TC PIP. It is not a protocol where you can say, hey, I want your attention, listen to me, I'm telling you something.

The controllers on that network are -- use what's called ethernet global data, which means they are programmed to look for things on the network, but can't be forced to.

In other words, their application code determines what it is they look at. Somebody else can't come in and say, listen to me. They will ignore that. That obviously has cyber security and network security implications.

The other thing associated with that is although we could -- I use the word control network. The controllers are set up. We have many, many dual and triple redundant controllers, and they are set up such that the controller has remote data acquisition in and out suitable for its function connected to it.

So, in other words, it doesn't have to ask another controller for what reactor level is in order to control level. That controller measures level itself, does the algorithms, and outputs to the feed pumps all on its own little -- I'm using the word network, but that's not the right way to phrase it.

But the idea is it's autonomous. So, in other words, if this whole network was somehow flooded with the traditional data storms or spoofing or bad guys, these controllers continue to work autonomously. They do not need the network.

Okay. So what we do use it for, you know, is to provide operator inputs to things, and of course for the controllers to send out information so that the operators can see.

But the important thing is that if they don't get any control inputs, they'll just operate at their last known values.

It also happens to be -- these are -- we call them managed network switches. They have far more capabilities than a traditional ethernet switch.

So, for example, you might imagine a switch which has, you know, 25 ports on it, and we determine what gets plugged into the port and set up the switch to say, okay, this is your configuration. And so when somebody comes along with a laptop and plugs into a port, the switch says, I don't know you, I'm not going to listen to you.

Okay. So -- and other things that it does -- I can't get too far into cyber security in an open meeting, but other things it does is it recognizes traffic on its ports.

So, in other words, if a controller decided to go completely berserk and flood that switch, the switch will turn off the port.

And since everything is dual ported to two switches, the processes still continue. So, in other words, it's an ethernet switch that has all the traditional functions of switches but has a lot more functions in terms of what you might call network security and network management.

And, of course, all these things that when the switch determines that somebody is trying to do something bad, the switch will of course end up with an alarm in front of the operator, so that it won't be silent, that this is happening.

So we have the controllers that very well protect themselves from being talked to. We have the switches that very well protect the controllers. We have these bridging work stations between the two networks, and then we have what you consider to be the traditional firewall which is much more than a traditional firewall, but without getting into great detail now, this is where the outside world is.

So the firewall is set up to be two computers, two redundant computers, with a shared memory between them.

So on this side of the plant, the internal computers scarf up everything, whatever there is to know about the plant, and dump it into a shared memory.

So the only function of that computer is to write into shared memory, and that is a very easy function to verify.

It of course has no ability or programming to read from the shared memory; it just dumps in.

On the other side of the shared memory are the external processors. The external processors of course can only read the shared memory but don't write to it. And even if they did, of course, the internal ones would ignore it.

So from the point of view of the external processors, the data appears by magic in the shared memory locations. Here's reactor level, here's reactor pressure. I have no idea how those numbers got there. I didn't ask for them, I don't know the addresses of any internal stuff in the plant to even begin to ask for them.

If somebody completedly spoofed me and said, hey, shut me down, they still can't get through the shared memory into the plant to control stuff.

The other important thing to say about that concept is we don't have any other ports into the plant. So, you know, there's not places for business networks and stuff like that to plug into the plant control system.

So basically we allow no external input from the outside world, and what the outside world sees from us is what we choose to put across the shared memory.

MEMBER BANERJEE: What are the external functions that you're --

MR. POPPEL: Technical support center, emergency offsite facilities, nuclear data link, the utility's business network, or the utility's engineering network.

So they -- now the external computers have like a traditional firewall function, so they will be programmed to say this is a good guy who is allowed to talk to me, and this is the kind of data we're allowed to send across.

However, that computer only knows what's in the shared memory. It has no ability to query the inside computers in the plant. Okay.

So if we don't send it over, nobody gets it, and we don't accept any data from somebody coming in. Okay.

We believe that to be -- and, of course, if they did get here, then they would be on this network where the managed switches would basically be saying, who are you? I'm not going to let you talk to anybody.

So the point of all of this is that cyber security and network security isn't a box we put in there. It's baked into the system all the way through.

And so in that context, the data isolaters that go from safety to nonsafety are in fact just part of that cyber security. But of course the software and hardware for that is safety related, where the other stuff is nonsafety related, but the function is the data doesn't come backwards through the system and it gets harder and harder and harder as you go further in to do stuff.

And, of course, the safety systems are organized into the reactor trip system, the ECCS systems, and we have a special more or less not programmable hardware-software platform for ATWS/SLC and vacuum breaker isolation function. They are not -- they do not use multiplexing, they are hard wired, et cetera, but they are safety related.

We have -- the system is deterministic. We had some discussion about that yesterday, but that should be understood to mean it's not event driven, it's time driven. I'm going to look at reactor level every X milliseconds, I don't care what else is going on, that's how often I'm going to look at it in order to make a decision as to whether or not to trip. That's fairly standard stuff.

We have four divisions, and the four divisions are used because we have -- the only data that is shared between divisions is that data needed to support two out of four voting.

And so all initiations, scrams, are any two unbypassed light parameters will cause the trip. So -- or we could send tons of information over the intradivisional networks. We don't. What we send is the trip status, the bypass status, and whatever we need to do message authentication, say this is a legitimate message coming from the other division.

Other than that, the divisions are completely autonomous, they stand alone, they do not require any backwards information from nonsafety in order to do their functions.

In fact, the nonsafety part of the system could be not there, it could be broken, the fiber could be disconnected, it doesn't matter. The safety systems will still work autonomously.

The other thing is that our reactor trip platform -- by platform I mean a hardware and software operating system, chassis -- is different from the ECCS. So they are not using the same operating systems. They are not using the same hardware. They are separate.

That carries through to the transmitters, sensors, whatever you -- so, in other words, if ECCS did one needs reactor level and reactor trip needs did one needs reactor level, two separate transmitters.

So the two safety systems, if you will, are not interconnected in the stuff that they have to do. They can do all their safety functions by themselves.

MEMBER BANERJEE: Are the electronic components, diverse in design and manufacture as well, or --

MR. POPPEL: Yes. Our product for the reactor trip and neutron monitoring system is NuMAC.

You may have heard the name. It's both a retrofit and is on the AWBR.

SSLC/ESF, which is the ESF function, is being done with Tricon, the Triconic system. That's a triply redundant platform per division, and the neutron monitoring and reactor trip stuff is NuMAC, no common operating systems, no common hardware. Okay.

And that, incidentally, carries through -- we'll see that when we get to the diversity chart.

MEMBER STETKAR: I was just answering Sanjoy.

You answered relative to the digital platforms. Since you have large numbers of input transducers, transmitters, whatever you call them, the analog input devices -- for example, you mentioned reactor vessel level, there might be 12 levels transmitters for the reactor protection SSLC/ESF and the diverse protection system, for example, or the nonsafety equipment.

Are those sensors also diverse in the sense -- from different manufacturers or are they same, essentially the same equipment? Are all 12 of them --

MR. POPPEL: No. The BWR as a fleet has talked about that issue before. But essentially they all use the same type of differential pressure technology.

MEMBER STETKAR: Same manufacturers?

MR. POPPEL: They don't have to be, but we didn't see any diversity gains in making them different manufacturers.

MEMBER BLEY: The last time we looked at that, some other manufacturers sold devices, but they had the Rosemont internals.

MEMBER STETKAR: Okay. I didn't want to get down to a specific. Thanks.

MR. POPPEL: Where we can do things -- like, for example, when we measure temperature, say, for the feedwater temperature trip -- I don't want get into too detailed -- the safety systems use thermocouples and the nonsafety systems use RTDs. So some places it's easy, and we do do that.

MEMBER STETKAR: I was more concerned about, you know, the levels, pressures, flows, that kind of stuff.

MR. POPPEL: Yes.

MEMBER BANERJEE: All of those measurements are made how, for the flows?

MR. POPPEL: We don't have any -- yes, we do have safety-related flow. They are also made by differential pressure measurements.

MEMBER BANERJEE: Orifices or venturis or something?

MR. POPPEL: Yes, orifice venturis, yes.

This also applies to the nonsafety systems, but I just want to show the example.

All of our DCIS components have redundant power supplies, and the whole component will work off of either. Each component has two power feeds, and the two power feeds each have their own inverter and battery systems.

So, in other words, we can lose one battery, one inverter, one AC power feed, and the entire system keeps working.

One of the reasons this was done is -- I'll say in a minute. But the important thing is it supports the self-diagnostics because obviously if the problem was power, and you only had one of them, you wouldn't hear too much from the system.

So we end up with, you know -- we have to ability to pretty much diagnose almost anything that happens in that design.

Another thing that is very unusual about a passive plant compared to an active plant, this plant was designed to be N-2.

Basically what that means is we can deliberately have a safety division out of service, completely out of service, and we can take a single random failure and we will have two divisions left that will operate all the ECCS, not just Div. 1, Div. 2 of the ECCS.

Obviously you can't do that with a motor, but what we have, for example, is all of our safety valves are either explosive squib valves or solenoid valves, or air-operated solenoid valve.

So we typically have multiple divisions on that valve, so that, in other words, what I'm talking now is DCIS, not mechanical.

We do have analyses for mechanical valve failures, but assume that the valve works, okay, it can work from division 1, division 2, division 3, and in a diverse protection system that I'll dicuss a little bit later.

So the same valve can be operated by any one of four systems. And that's how we can say that all of the ECCS works even though two divisions are completely gone.

Obviously that only works with a passive plant and, you know, not anything like active motors and pumps like that.

We mentioned that the safety systems are diverse amongst themselves. The safety systems are also -- this may be a hard thing to see. The safety systems here are meant to -- this is meant to indicate that they are diverse from the nonsafety systems, so the nonsafety systems have a different hardware-software platform than the safety systems.

This one is meant to indicate -- I'll talk a little bit about the severe accident deluge system, but the bottom line is the thing that operates the deluge system is in fact diverse from safety and nonsafety.

Then on the top line -- oh, also still on the middle line, a new control system for the ESBWR that has not previously been on any other BWR is the diverse protection system that is a nonsafety system that has been added to address common cause failure of the safety systems.

That box has the ability to scram the plant, isolate the plant, and ECCS functions, a subset of them.

But the bottom line is that's this, and it's diverse from these. So by definition it ought to be common-cause-failure proof for failures of the safety system. The thing that's addressing that has to be diverse from the safety systems. Okay.

The nonsafety system, I had already mentioned to you is divided up into the five segments. Okay. The nonsafety system, even though it's nonsafety, is in two different DCIS rooms that are not the same rooms as the QDCIS rooms, and those rooms are in different fire zones, and those rooms are separate from the main control room fire zone.

So, in other words, all of our DCIS back panel rooms are fire zone physically separate from each other and from the main control room, and from the remote shutdown panel.

We had a -- oh. I had mentioned the fire zones. The nonsafety DCIS, like the safety, is powered with two or three uninterruptible power supplies, so it has the same robustness and power as do the safety systems, and also the same diagnostic capabilities if power goes away.

I mentioned we don't use the networks for closed loop control, and the components are diverse.

The diverse protection system, this is a little more detail about that, is the way the diverse protection system works is we analyze the chapter 15 events, accidents and transients, assume the safety systems have suffered a common-cause failure, and then compare the results to 10 CFR 100.

Should that not be successful, we add the function to the DPS system. So the DPS has a back-up scram function. It doesn't have anticipatory scrams like say control valve fast closure because if you didn't have a control valve fast closure, you will surely go out on flux or pressure with just a slight delay. The delay is fine with 10 CFR 100 limits. Okay.

MEMBER APOSTOLAKIS: Would you explain, please, what you mean by common-cause failure?

MR. POPPEL: It's not well defined except to say that for some reason all of the safety systems simultaneously stop working. They will not perform their functions. They will not scram and they will not isolate, and they will not initiate ECCS.

MEMBER APOSTOLAKIS: Based on software?

MR. POPPEL: Because something in the system, even though these divisions are asynchronous between them is, as I described, the assumption is they're gone.

MEMBER APOSTOLAKIS: So you are not really looking for the causes of the common-cause failure, you are saying if it happens, I have a way of managing it?

MR. POPPEL: We already tried the argument about it ain't going to happen, and that's not the answer. The answer is it happened.

MEMBER APOSTOLAKIS: You are managing the failure due to some cause.

MR. POPPEL: Yes.

MEMBER APOSTOLAKIS: And you are not exploring what the cause might be.

MR. POPPEL: Well, I mean obviously if we found something that could cause a common-cause failure, it would never make it through our software QA, obviously.

But the point is we didn't cover everything. The important thing is the diverse protection system, if it needs to scram on level, it has its own level and it's not the same level as the safety system transmitters.

MEMBER CORRADINI: Just to say that a different way, so this is in itself its own division. It has its own sensor, its own controller, its own feedback to this, so it's just another division.

MR. POPPEL: Yes, but it's a very special division in this way: It's a triply redundant controller. Because it's nonsafety and because we want it to be very reliable, as do our utilities in terms of not inadvertently doing things, okay, nonsafety is allowed to talk to each other, so you get a lot more robust control system if you have triple redundancy, and the controllers can diagnose themselves.

We also don't make it fail-safe, unlike the reactor protection system, because it's a back-up to a back-up to a back-up, and it doesn't have one level transmitter, it has four.

So the three -- the triply redundant controllers are each doing two out of four logic for say reactor level, reactor pressure, suppression pool temperature, et cetera.

MEMBER CORRADINI: So the different sensors, four of them, and they're voting for each of the three controllers?

MR. POPPEL: Well, each of the controllers sees all four sensors.

MEMBER CORRADINI: I'm sorry, I'm sorry. You said it better.

MR. POPPEL: But the important concept -- I mean we're making it very reliable to scram and very reliable to not scram inadvertently. But the point is that hardware-software platform and those transmitters and stuff are separate and get to the actuators in a separate path, and in other words, they don't tell the reactor protection system to scram, they separately interrupt con to the scram solenoids.

So, in other words, we don't assume anything that is working in the safety systems.

MEMBER CORRADINI: Okay.

MR. POPPEL: Now, as I said, it's a subset. We don't isolate and scram on everything, but what we do isolate and scram in ECCS on is enough to get us through the 10 CFR 100 limits.

MEMBER CORRADINI: Can I ask an opposite question? So you said you used the 10 CFR 100 limits. What things did you not need to have a back-up for?

MR. POPPEL: So far that's an -- because early analyses are not yet done, I can't answer that, but for example, one thing I can answer you is the example I just gave about the reactor will typically scram on a turbine trip, it will measure the stop valve, you know, fast closure and scram.

MEMBER CORRADINI: And you decided that wasn't necessary?

MR. POPPEL: Right. Because that's an anticipatory scram in the real reactor protection system, but if you never had it, you would go out on pressure or flux.

So basically what happens is you get a scram, it just has a longer delay in it than -- delayed meaning, you know, tens of milliseconds as opposed -- before you actually get the scram from the DPS system.

MEMBER BROWN: Just to reiterate and emphasize one point, Michael, that you brought out, is that all four sensors feed all three of the triply redundant controls. In the reactor protection system, there's four water level sensors also, but it's only one feeds each division.

So I just wanted to explicitly say what that difference was, for the pressure level, nuclear, whatever, ad nauseum.

MR. POPPEL: The IEEE-603 independence of divisions wouldn't let us do the things that a single triply redundant controller could do.

MEMBER CORRADINI: Got it.

MR. POPPEL: Okay.

MEMBER MAYNARD: I think you covered this, but I wanted to make sure I understand. On the diverse system, it will actuate if the safety system fails to. Is there any way that -- there isn't any communication between the two systems, though, is there, where the safety system can't tell the diverse system to not work?

MR. POPPEL: That is correct. And a specific example of say the reactor scram, you may recall that the boiling water reactor has two scram solenoids, and they are energized to not scram. So the reactor protection system has all the fancy load drivers and stuff, if you will, in the 120-volt supply to those solenoids, and then the diverse protection system has a switch in the 120-volt return from those solenoids.

So neither one can prevent the other from scramming, okay, and but of course that means either one could scram you, so we want to make sure it's reliable in terms of not scramming you when you don't have to scram.

MEMBER BROWN: Relative to the diverse protection system, to the normal four divisions reactor protection system, do you try to provide some difference in set points so that the diverse protection system wouldn't sense that it would need to scram and the regular protection system would say, hey, I'm happy? They're higher or lower, whatever the --

MR. POPPEL: There has been a lively discussion about that in GE. One of the things you should know is because of other jobs that the DPS does in terms of ATWS and things like that, there is a data link from the safety to the diverse protection system. So the diverse protection system knows everything the safety systems know -- not vice versa.

So one of the things they decided to do is have the diverse protection system scram for any of its functions, or if the reactor protection system scrams.

MEMBER BROWN: It's a series.

MR. POPPEL: And so -- well --

MEMBER BROWN: I mean it's being told --

MR. POPPEL: -- it's an organ.

MEMBER BROWN: Yes, I got it.

MR. POPPEL: It doesn't need permission to scram, but it will scram, because this way people wanted to make sure that if you had a scram in one, you'd have a scram in the other. That's sort of a human factors thing. The data link is there to be used and it's the specific programming of the DPS that --

MEMBER BROWN: So if you had similar set points, you had the exact same set points in each system, the diverse protection system could scram you?

MR. POPPEL: Right.

MEMBER BROWN: Independent and ahead of, theoretically --

MR. POPPEL: Theoretically.

MEMBER BROWN: -- ahead of the reactor protection system. Depending on tolerances, blah, blah, blah. Their detectors, what they're feeding in relative to the other systems.

MR. POPPEL: And not only tolerances, in addition, the cycle times of the controllers. I mean it take 10 milliseconds more to get to a scram decision in this controller than in that controller. Okay.

In general, the DPS will probably be faster because it is synchronized between those three control channels, whereas the reactor protection system channels are completely independent and asynchronous, so you could imagine at the two out of four input gate, this one said reactor level and just missed it, where this one said it was there. Because their 25 milliseconds of looking at reactor level aren't the same 25 milliseconds for each division.

MEMBER BROWN: Got it.

MEMBER BLEY: Ira.

MR. POPPEL: Yes.

MEMBER BLEY: What kind of separation

requirements are there for the data and control cables from the diverse system to the others?

MR. POPPEL: In fact, their requirements are that they not be in -- well, first of all, all of the safety equipment, just like a traditional plant, has its own cable trays, conduits, duct banks, et cetera, and in their own fire zones.

And the DPS system, in general, because the reactor building is divided into four quadrants, each of which is a safety division, but there's also nonsafety equipment in there, including DPS nonsafety equipment. So the DPS nonsafety equipment is always located in at least -- in other words, if you will, the four sensors and remote multiplexing equipment is cut in half, and half of it is always in a separate fire zone, okay. And it doesn't use any common cable trays or -- and is the appropriate Reg Guide 175 away from cable trays with other equipment in it.

The DPS system, like the reactor protection system, will scram on any two like unbypassed parameters. So, in other words, if half of the remote multiplexers for DPS in the reactor building go down, it's still capable of performing a scram or an isolation. Okay.

So it's a very robust, redundant system

all by itself, just implemented differently. And, of course, it has to be implemented differently because it's common cause.

MR. KIMURA: Ira.

MR. POPPEL: Yes.

MR. KIMURA: The rest of the -- the DPS controller itself has to be in a separate fire area from the nonsafety as well?

MR. POPPEL: I should have said that, yes.

MEMBER BLEY: But the cables have to run through some of the same fire areas as the other?

MR. MILLER: They are going to be in conduit so they'll be a separation.

MEMBER BLEY: The conduits are a requirement?

MR. POPPEL: Yes. The -- this is like the same thing as the cover slide. This is what the main control room looks like. A few points to mention -- well, this is what the draft main control room looks like.

Most of our stuff that interfaces with an operator has a very formal HFE human factors process to go through in terms of what they see, what alarms they see, how they actually control stuff, and that process is just starting. It is by no means complete.

What we have done is given them a backbone architecture and stuff to do all of that stuff on.

So here are four divisional displays -- we have one, two, three, four -- and here are four divisional displays. So we have two displays per division in the main control room.

The only way to talk to division one to tell it to do anything is with a division one display. It cannot be done from nonsafety, ever. No backwards control from nonsafety into it.

Additionally, if you want to do something in Div. 2, you cannot do it from a Div. 1 display. You must go to a Div. 2 display to cause that.

So, in other words, when you think about those solenoids, for example, or those squib igniters and a manual initiation, if you want to fire the Div. 2 initiator, you're going to do it at the Div. 2 VDU, and that's the only place you can do it.

MEMBER CORRADINI: I'm sorry. May I? Just to follow that, so -- because you said it earlier when you were just kind of in a general overview. And then if you didn't do it in the main control room, you have another location --

MR. POPPEL: Yes.

MEMBER CORRADINI: -- that again has the

same sort of logic separation.

MR. POPPEL: Yes. One of the things that's also -- the traditional remote shutdown panels available to plants are a very, very small subset of what is available in terms of control. And they have all kinds of transfer switches, they have all kinds of interesting problems associated with fire zones and divisions in the panel.

We have two remote shutdown panels in separate fire zones from the main control room and the DCIS rooms.

Each remote shutdown panel has a Div. 1 and a Div 2. VDUI wherefrom you can do anything in Div. 1 and Div. 2 that you can do from the main control room. Not a subset; anything.

In addition, it has two displays connected, one to the plant PIP A, plant investment protection, A network, and one to the plant investment protection B network.

The result of this is so four displays and it has a manual scram switch and it has a manual isolation switch.

Those switches are software free, just like in the main control room. So if you push those, you are going to scram; no software. Okay.

In terms of the displays, if you have offsite power available, either preferred or alternate, you can run the balance of plant, PIP, RTNSS, and safety.

If you only have diesel power available, you can't run the balance of plant. You can run the displays, but of course you can't power any of the equipment, but you can power the FAPCS, CRD pumps, you know, all of the plant investment protection things.

And then if the diesels are not available, you can run any of the Div. 1, Div. 2 safety stuff from the 72-hour batteries of those divisions.

And since they are connected to the -- the remote shutdown panels do not run through the main control room. They connect in parallel to those back panel areas appropriate to the division or nondivision via fiber optic cable.

We also had a lively discussion yesterday about inadvertent actuation. Our belief is that a communication message authentication and robustness is such that it will be highly unlikely that it would ever -- that any stress in the main control room from fire or smoke would cause an inadvertent actuation.

We can talk about that in some detail later, but the bottom line is, the main control room equipment being in trouble is not going to affect the automatic or manual capabilities of the safety or nonsafety systems because they are not in the same environment.

We have -- these are the nonsafety displays -- oh, I'm sorry.

MEMBER STETKAR: Can I back up to the safety for just a moment? I want to get something straight in my mind because you mentioned that got me thinking.

You said that only -- if I go to a Div. 1 VDU, safety-related VDU, and I want to actuate ICS, I can only actuate the Div. 1 signals for ICS from that VDU. Is --

MR. POPPEL: Correct.

MEMBER STETKAR: -- that what you said. Which means that to fully actuate ICS, I imagine that I must do it from two different panels because of the way that the signals -- I don't know how the signals are distributed among all the valves, but I did a little bit of reading in much detail for this meeting, but I think that means to fully actuate ICS, to get all of the ICS working, I need to do it from at least two VDUs. Is that correct?

MR. POPPEL: That is the present design,

yes.

MEMBER STETKAR: Thanks. That helps.

MR. MILLER: Just to reiterate, in the control room there's two Div. 1 VDUs, so you have redundancy in case one --

MEMBER STETKAR: Yes. No, I just wanted to get the fact that I couldn't go to one place and say start ICS for everything.

MR. POPPEL: And as we discussed yesterday, one of the human factors things that isn't going to change is the operator always has to do two things to get any action -- you know, select and fire; arm, fire. You know --

MEMBER STETKAR: But that would be from the --

MR. POPPEL: From the VDU.

MEMBER STETKAR: Yes.

MR. POPPEL: So that one single thing will not -- you know, spilling the coffee, putting your elbow on the display, will not cause something to happen.

MEMBER MAYNARD: But for some of the valves that are like the squib valves, initating from Div. 1 is going to fire all of them; right?

MEMBER STETKAR: No. No, it won't.

That's my point. It will fire a subset of them. You will fire one-fourth of them.

MR. POPPEL: We have -- here's the way that is getting discussed in San Jose.

If the manual initiation for the division goes in, if you will, ahead of the two out of four logic, then your statement is correct, you need to have say any two divisions where they say manual.

If it goes in after the two out of four logic, then it will -- if you blow the Div. 1 thing. And so the human factors folks are having a debate about which is the best thing to do.

MEMBER BROWN: I didn't get that. Repeat that.

MR. POPPEL: Forget manual. All of the automatic actuation of the say the isolation condenser, say on reactor level. So there's four divisions. They each look at reactor level, they each determine that there has been a reactor level initiation, and then each division sends that information to all the other divisions.

So at the input to the two out of four logic, there is a Div. 1, Div. 2, Div. 3, Div. 4 decision that says you should initiate, and the two out of four logic does that.

Then the output of the two out of four logic blows the Div. 1 squib. The output of the Div. 2 two out of four logic blows the Div. 2 squib.

So this debate for manual is am I going to put the manual signal ahead of the two out of four logic or after the two out of four logic.

And, you know, for everybody who argues that, well, you know, if you put it ahead, that means the operator will really have to think about it and do two things, and the other half says, well, no, no, no, but then you're forcing them to do two things in a stressful situation.

We always need an odd number of people to solve human factors discussions.

(Laughter.)

MEMBER STETKAR: You said that wrong. You said an odd number of people. How about a number of odd people?

(Laughter.)

MR. POPPEL: In any case, though, I mean, in other words, the discussion we're having is a question of the logic. It's not a question about the hardware or the software of the system. It's just how they want to implement it.

In the fullness of time there will be

detailed logic diagrams that in fact will show you it's before or after that, and can be commented on.

MEMBER BROWN: In light of that, can the reactor protection system for manual -- forget all the other interrelations, but the operator decides to scram, how many actions does he have to take to scram all --

MR. POPPEL: TR is literally about 10 different ways to scram the plant, but this -- in the case you're asking --

MEMBER BROWN: I'm talking about the --

MR. POPPEL: -- we have the traditional two big red push buttons. If he pushes both simultaneously, he will be directly interrupting, without software, the con to the scram solenoid.

MEMBER BROWN: You said that yesterday. We didn't ask the specific question, I didn't ask about how many --

MR. POPPEL: But it's also possible --

MEMBER BROWN: But there's two --

MR. POPPEL: You can barely --

MEMBER BROWN: And they are not -- they are strictly standard switches?

MR. POPPEL: Yes. There's no software --

MEMBER BROWN: Hardware? Okay.

MR. POPPEL: No fiber optics, no software. It basically winds its way down to a contacter and the contacter -- normally closed contacter -- and the contacter opens up the thing.

In other words, the contacter is in series with the fancy load drivers.

MR. KIMURA: And the remote shutdown system panels have the same buttons?

MR. POPPEL: Yes.

MEMBER BLEY: Rich, you pointed to one place. Is there only one place where those two buttons are?

MR. POPPEL: No, they are also in each remote shutdown panel.

MEMBER BLEY: But there's not two sets of them ?

MR. MILLER: Just one set. Just one set.

MEMBER BLEY: Okay.

MR. POPPEL: However, okay, I said the DPS system can scram them automatically. They have determined that it is useful to have the DPS system be able to scram you manually. Okay. That's not software free, but in other words, the operator, if he really wants to scram, and he's pushed the RPS buttons and for some reason it didn't scram, he can always push the DPS buttons.

MEMBER BROWN: Are both on his panel?

MR. POPPEL: Yes.

MEMBER BROWN: You don't have to go to the panels to do that?

MR. POPPEL: No. But you can go to the VDUs and put in scrams, you know, like -- in other words, you can do a two out of four reactor scram at the VDU. Insert a trip in this division, and if you do it in two divisions.

In addition, you guys know about ATWS and SLC to shut down. It may -- I don't know how familiar you are with the ABWR, but these rods are not -- control rod drives are not the traditional latching piston. They are motor drives and a hydraulic scram.

So the scram is what you're used to, but the normal positioning is a motor, not a hydraulic latching piston drive.

And so if you will, there is a nonsafety motor scram. In other words, the system says, oh, I got to scram? I don't care whether it's scrammed hydraulically or not, I don't care what position the rods are being told to go to, you are now being commanded to go all in.

MEMBER BLEY: You mentioned that

yesterday. How fast is that?

MR. POPPEL: It takes about two minutes.

MEMBER BLEY: Okay.

MEMBER ARMIJO: Compared to the hydraulic is seconds?

MR. POPPEL: Yes, two or three seconds, the traditional time you're used to.

MEMBER BROWN: I would take it that the drive-in is not necessarily fast enough to provide a safe reactor shutdown? Or is it?

MR. POPPEL: There are analyses which show that if you can scram in two, three minutes, you'll be, if you will, but obviously it's not going to be at 10 CFR 50 stuff.

MEMBER BROWN: One other question. You think you can get the red button to scram. What about the ECCS ESF type functions? If you wanted to manually initate those, can you do those manually, bypassing all the software as well?

MR. POPPEL: You can, and you do it through the VDUs. This has been another human factors debate.

MEMBER BROWN: But that's not --

MR. POPPEL: That's not software free. We wanted the software in the loop because what you're doing, for example, when you open up the pressurization valves, is you are initating a LOCA. And so, therefore, don't do it unless you really have to.

And so therefore we want to make sure that the operator is well supervised. On the other hand, both DPS and the safety systems can in fact operate that system.

They also have -- if you look at our aaes, the depressurization sequence of the reactor starts with the safety relief valves as opposed to the pressurization valves.

There is a sequence blowdown, I guess is the right way to phrase it, to get you to the pressure where the gravity drain pool systems work. Okay. And it's deliberately a very long sequence. Okay.

So there is no particular reason the operator has to do anything fast for when this happens. It takes a while to get to level one with this plant.

MEMBER BROWN: One more question. The VDUs are all touch screen?

MR. POPPEL: The safety VDUs are touch screen. The human factors people are debating whether or not the nonsafety ones should be touch screen.

The DPS ones will be touch screen. Okay. Just so it will have the same sort of interface as the safety ones, even though they -- incidentally, I should say these aren't the same physical displays. The safety displays have an operating system and a physical display that's different than the nonsafety.

So, in other words, you can't have a common-cause failure of the VDUs, either. You can't have a common-cause manual failure just like you can't have a common-cause automatic failure because of the DPS and the two different ways to initiate.

MEMBER BLEY: Ira, I may be mixing up different designs in my head right now, but in this plant, can the operators locally -- or, you know, local at breakers and contacters -- start pumps and change valve positions or do they have to do that through the DPS or through the control system?

MR. POPPEL: Well, normally you do it through the control room, but the MCCs and switch gear have manual operation capability on them, if for no other reason to help you when you restart -- I'm sorry, motor control center, or things that operate valves.

That's not a relevant question to the safety because we don't have motor-operated valves in the safety systems, but it is a relevant question to the plant investment protection and stuff like that.

So you can operate those things locally if you -- should you have to.

Finally --

MEMBER BROWN: One more question on the VDU touch screen issue. Is the human factors aspects of inadvertent operation on the touch screens?

MR. POPPEL: Well, that's one of the reasons we have the two actions. So, in other words, in Lungman -- the reason I sound so hesitant is because this is really a big HFE deal, but for example, in Lungman, you can imagine looking at a P&ID display of feedwater or something, and then you would touch a valve or a pump, and what you get is a pop-up that basically says you want to turn it off, you want to turn it off.

If it's an analog positioning thing, you can see the position, you can set the set point demand moving.

So, in other words, the operator has to select it, one action, and then he can do something with it. But he can't -- you know, you need the two things to cause the action to happen.

So a single VDU touch doesn't do anything, in every system.

MR. MILLER: Was that question?

MEMBER BROWN: That's one of them. The other was for -- I mean you've got your big wide channel display to give you a general picture of the whole plant. Normally the screens on the VDUs are somewhat smaller, and so you would have subsets or different screens, different displays that you would want to call up.

Is that -- is every screen that you would want to view -- and I view that as a screen selection that has a number of displays on it, whatever they are, that you predesigned to have certain types of information. Do you have to menu-select those?

MR. POPPEL: Yes.

MEMBER BROWN: In other words, you call up the menu, drive -- whatever you do with the thing --

MR. MILLER: You can do a menu, or you can do --

MEMBER BROWN: It's a function of separate -- as opposed to a separate touch which appplies to every one of say six or seven or eight screens?

MR. POPPEL: Well, actually, there are hundreds and hundreds of screen formats that accomplish various things. But when we say menu --

for example, what we did in Lungman is you can imagine a main menu where it says, you know, here's the reactor and here's the safety systems. I select a safety system and then I get -- you want to see a P&ID, you want to see alarms, you want to see this, that, or the other thing.

In addition, because it's a screen it's very hard to fit, you know, large systems on one screen, and so one of the things you have on the Lungman menus is on the screen you can have -- see where this pipe runs off the screen? If you touch the pipe there, there will be a little arrow, and you can get to the screen that it connects to.

And you can also do things like let me see the previous screen you just looked at. Okay.

So there's other ways than the hierarchical main menu of getting through to things that are organized by the HFE group in hopefully an intelligible human-friendly way.

Additionally, I didn't say it yet, we didn't talk about the wide display panel, but one of the -- this is foreshortened, but one of the features of that panel is an operator can select any display he's looking at and put it up there.

So, in other words, if you said this is

really interesting, everybody should see this, and he can do that. So can the shift supervisor. Okay.

MEMBER BROWN: It doesn't take up the whole display?

MR. POPPEL: No, no, it's not -- so, in other -- basically the human factors rules are -- there are some signals that are so important and so useful or so, you know, entry conditions to emergency procedure guidelines, that we believe that even though it's a display that they should be fixed. So the operator level is always there, because that's really important, and he really needs to see that.

And, of course, the HFE debate is what signals are those. Everybody would agree on level, but there's other signals. And so you end up with like, if you will, a fixed mimic of the vessel with a water level moving up and down with reference to level one, two, three, and the core, and stuff like that, and it's a very easy picture.

Plus the screen itself indicates that what I'm showing you, at least transducers have agreed on. So the operator, you know, gets his rule, like don't do anything unless you get two things to agree. And so that's important.

In general, we have the electrical system, which is also important, and then we have the variable display. But the idea is that it's human factored, and it has to be decided.

The physical -- the human factor parts that everybody does agree with is the screen -- the variable display begins here, so everybody in the control room can see it. From every place in the control room in that line, meaning the shift supervisor, everybody has got a line of sight, and the letters and numbers on it are sized so that you can read them from that position. So, in other words, it's not like six-point type from 35 feet away. And so that's important, is everybody has sight lines to it.

There's other "golly, gee whiz" stuff like there's displays here where the operator can, you know, call up P&IDs for the plant.

In other words, you know, like a tech storage. These panels are also here to support surveillance activities --

MEMBER CORRADINI: There's a question.

MR. POPPEL: Oh, I'm sorry.

MEMBER APOSTOLAKIS: No, finish your --

MR. POPPEL: I'm saying they're there to support surveillance activities, basically so they're not standing in front of the operator or standing in front of the wide display panel, you know, because he's going to be there for a while doing stuff, so he can do it over there, still be supervised, but nevertheless, you know, not interrupt the control room.

MEMBER APOSTOLAKIS: Is the thrust of your question, Charlie, to explore what could go wrong? Is that really what you're asking?

MEMBER BROWN: Well, I am just addressing an issue we addressed similar to when we started implementing this stuff. The same debate as to how did you select and put information in front of the operators in terms of what they would need normally for their operations.

And I guess we have some menu items that they were relegated to -- they were the maintenance items, maintenance screens, stuff like that, where your plant is shut down and we determined and figured that there were half a dozen screens that illustrated what would the operator want to have at his rapid beck and call.

MEMBER APOSTOLAKIS: And this is not what's being done here?

MEMBER BROWN: Well, I don't know.

MS. CUBBAGE: Excuse me. This is Amy Cubbage, NRR staff.

I think it's a fascinating topic, but it's really not the topic we're here to discuss today. The human factors interface is being covered under a different chapter, and we'd be happy to get back to you if you want any additional information about the design process on that. Or GE would, because that's their responsibility. But I think we need to move on if you want to hear from the staff.

MEMBER BROWN: I thought we had these guys until 9:45. So I thought we --

MEMBER CORRADINI: Well, we don't want to capture the staff only for 15 minutes. We want to give --

MR. MILLER: I think we're finished. There's only one slide.

MS. CUBBAGE: I didn't mean to cut you off, it's just that that wasn't the topic.

MR. POPPEL: But, as Amy said basically, this part of the presentation --

MEMBER BROWN: Well, but the reason I asked the question is to say -- okay, not necessarily it would develop it to data flow. How much did they need, how was it going to be relevant to some of our comments, because there were questions yesterday.

MS. CUBBAGE: Okay.

MEMBER BROWN: That was the fundamental thrust -- you asked, okay, was it -- depending on how you did that, then you had different data flow requirements, and so that's one of the reasons I was asking.

MEMBER CORRADINI: You missed the joy of January. We had that fun in January.

MEMBER BROWN: Which one, the HFE one?

MEMBER CORRADINI: Yes.

MEMBER BLEY: We're getting a bit more information now than we got.

MEMBER CORRADINI: I'm just observing that --

MEMBER BROWN: Oh, you mean what we've done?

MEMBER CORRADINI: Well, no, we had the first shot at --

MEMBER BROWN: I understood we weren't.

MEMBER CORRADINI: There was a lot of open items, just to clarify. So in chapter 18 -- did I get that right chapter -- we looked at chapter 18 then. There was a lot of open items, but specifically on human factors on how you display this, and we heard from Brookhaven as the staff's consultant -- is that the proper word -- to understand those open items at that time.

But you're right, there is more now than we had heard then.

MEMBER BROWN: Okay, we'll go on.

MEMBER ARMIJO: A quick question. Does something like this physically exist at Lungman right now?

MR. POPPEL: Yes. And at K-6, K-7.

MEMBER ARMIJO: So the heritage is the Kashiwazaki plant's ABWRs, and to the Lungman with some changes or improvements?

MR. POPPEL: The general layouts of the control rooms of all three of those plants is the same, although obviously the hardware, software, and there are those who think that the science of human factors has advanced over 15 or so years so that you probably wouldn't display the same things on this plant that you would in K-6, 7.

One of the things I should also --

MEMBER ARMIJO: Japanese.

MR. MILLER: We had started with those plants and progressed with lessons learned and so forth.

MEMBER ARMIJO: Sure.

MR. MILLER: And this rendering here is probably our fifth or sixth rendering on the ESBWRs, starting from the Lungman design, moving to new technology, and things of that nature, and being more specific to the ESBWR systems.

MR. POPPEL: We should also mention that this is an automated plant, okay. Without discussing, you know, the details of automation, the human factors of automation is to take burdens away from the operator, okay, so that he can be a supervisor of things happening rather than doing that.

So, for example, this reactor will pull itself critical, okay. I mean there's an automation sequence, and it says, okay, go to this break point, and it says pull critical, you push the button, okay, and the reactor will pull itself critical, and I mean it's interesting to pull a reactor critical, but after you have done it once, you've really had all the time you ever need to do it. You know, it's boring, and it's time consuming.

MEMBER BROWN: But these guys will never do it once.

MR. MILLER: They might want to do it manually.

MR. POPPEL: Well, I'm sure -- everything in the simulator, I'm sure will be manual as they train. But in the real plant, the intent is to take burdensome things that are repetitive.

As an aside, because somebody, I'm sure, will ask the question, all of the things that supervise the operator in terms of when he does it manually, rod blocks and neutron monitoring system blocks, that's all still there with automation.

So the automation system is just as supervised as the operator, and it's not supervised by itself. In other words, those same things that would block an operator rod pull will block an automation rod pull.

So, in other words, we're not -- automation isn't trying to get away with something. Automation is a tool to remove burdens from the operator. That's pretty much the way the safety systems are set up to respond to accidents also.

So the intent is that the operator should be able to just see what's happening, follow the sequences, and only have to intervene when something doesn't happen that should. Because in general, it will -- if it all works like it's supposed to, if any two divisions work like they're supposed to, the reactor will respond as analyzed.

MEMBER CORRADINI: Do we need to move on?

MR. POPPEL: Yes. This is the last slide, and I've pretty well covered it by mentioning all the stuff about the remote shutdown system, it's just basically a little control room.

MEMBER BLEY: So I want to follow up your last comment just a little, even though it's broaching into the human factors engineering.

What do you guys do in designing the system to keep the operator involved in that supervisory role since he's not actively involved in carrying out the steps?

MR. POPPEL: That one is straightforward. It's not like one big switch that you say here's the reactor cold iron, and when you're done, you're at 100 percent power.

Basically the automation system -- in Lungman it was about 35 steps, so in other words, to get the plant from cold metal to 100 percent power, there were 35 break points, you know, so de-aerate the reactor, pull it critical, heat up and pressurize the reactor and stop at 400 pounds. Rated pressure and temperature. Roll and synchronize the main turbine.

So, in other words, each time the operator said that, it went through that, and you could see it on the steps on the screen, and then it would stop and the operator has to give the next break point permission.

So, of course, he can stop the automation at any point, but the thought is that if we've done it right, at the end of a break point he can look around, look at the big screen, you know, does everything look fine, and then he can press the next button.

So he's forced to be involved periodically in the sequences.

MEMBER SIEBER: Do you actually roll out the turbine?

MR. POPPEL: Yes. Matter of fact, all of the stuff is -- the implementation is different, but like, for example, back at K-6, 7, all of this stuff is happening. The turbine roll, the reactor critical, you know, power is -- although it's not likely to happen in the U.S., for example, our customer contract requirements in Lungman require that the plant be remotely dispatched. Require, not allow.

And so, therefore, the grid operator can say -- there's all kinds of controls on this, as you might imagine -- Lungman should go to 900 megawatts. And it will. Okay. And the operator can do the same thing. Of course, he has to give permission for that to happen.

That's not -- obviously isn't going to happen in the States, but the capability is there in both Japan and in Lungman, and in our European ESBWRs.

MEMBER SIEBER: They have to have a license.

MR. POPPEL: In the United States.

MR. MILLER: Yes, in the United States you need a license.

MR. POPPEL: Okay. But there they want to treat the plants just like any other power plant on the grid.

MEMBER SIEBER: Right.

MR. POPPEL: Okay.

MEMBER CORRADINI: Thank you very much. I wanted to leave time for the staff and also so the members can ask questions of the staff and back to you, if necessary.

So is Mr. Li our lead staff member to start us off? Oh, I'm sorry. Excuse me.

(Pause.)

Dennis, are you going to start us off?

MR. JUNG: Very briefly.

MEMBER CORRADINI: Okay.

MR. JUNG: I apologize, I don't have the nametag with us. I'm Ian Jung, the chief of the instrumental and controls branch.

MEMBER CORRADINI: We recognize you.

(Laughter.)

MR. JUNG: We spent a lot of time yesterday.

Hulbert Li is the lead reviewer. We have a number of actually staff members involved.

As you can see, through our SER, which is about 240 pages long, it was a result of the significant staff effort to cover the I&C area.

With that, Dennis is the project manager. This is Dennis.

MR. GALVIN: Ian said most of what I wanted to say, but --

(Laughter.)

The slides cover the topics in the SER 2 through 6. The reviewers -- we had the applicable regulation and guidance, and we have about 70 RAIs open out of 276. Ian is going to go into the details of the review, and then our open items.

MR. JUNG: Thank you, Dennis.

I thank the committee for listening to us and providing insights and perspectives yesterday. It was quite useful, and we expect the same thing today.

Let's also take this opportunity to thank many of my staff projects and other action leaders, other divisions and branches, who worked together with also GEH supporting us in terms of addressing safety issues.

The bottom line is the staff used the chapter 7 of the SRP as the main staff guidance which references a lot of the regulatory guides, which also endorses many of the industry standards.

Fortunately, in our digital I&C, in all I&C in general, we kept up with our guidance document to accommodate digital I&C through the experience of the ABWR System 80-Plus and AP1000 and so on.

Are we up to the current state of knowledge in terms of the current guidance? Maybe there's always delta. We expect some delta all the time, you know, technologies developing, and field programmable gate arrays that might be in the new power plants. We may need additional guidance.

But there are industry good practices. We will try to use that information as we go through the detailed design process.

Just to summarize, GEH is the design certification submittal. As we have talked about through the chapter 14, maybe also human factors engineering chapter, GEH provided the design information at a certain level. Not all the information that is necessary and sufficient to force that to make the original finding. That's the bottom line.

And GEH opted to use the concept of a design acceptance criteria, which has been endorsed by the commission through several SECY papers in the '90s, and the more recent Part 52 change actually introduced word design acceptance criteria as an example, and though the expression has been first time introduced in the rule language.

So the use of the design acceptance criteria has been a commission policy, if not necessarily a rule. But we exercised the use of design acceptance criteria through ABWR and AP1000 in the past.

So for the staff going in, we wanted to follow and improve upon what we've done in the past. So what GEH has submitted, the high level of design information functional requirements, and also design acceptance criteria they had proposed. They are not -- they are consistent, even better than what we have seen in the past.

Yet on the steps, our initial desire is to have as much design details as possible so that we have a clarity in what the design looks like, and also in combination of DAC description that is consistent with the information policy, which is it is clear, and once it is followed, the staff will be able to verify it down the road without ambiguity and confusion or contention.

So the staff had two approaches, basically looking at GEH's submittal, make sure what they had provided, even though it's a high level and GEH just went through high level design, some of the architecture information, high level functional requirements. Some of what actually GEH said this morning, that some of the information is not in DCD. You recognize that, or will go through HFE process, or will develop the detailed logic diagram.

So we looked at what we were provided with in DCD, whether that be meeting the regulations that we review under, SRP and the GDC, and IEEE-603, which is endorsed by 50.55(a)(8), which is regulations.

That's one big area we focused on. A second area, of course, we looked significantly, we still are spending a lot of time on, a lot of the IAI question 70 are related to actually the second part related to the clarity in the tier 1 that you mentioned, design acceptance criteria, and ITAAC description. They are clear to the staff so that we can verify down the road.

We had a lot of discussion about the level of detail yesterday, and that's not unique. We have a lot of discussion internally among the staffs. Sometimes we have an odd number of people to make some decisions on what level of detail we needed, maybe even in the step side to it.

Some of it is subjective. You know, the guide has not been fully exercised to be at plant operational stage, so we are learning as we go at this stage.

So I want the committee to recognize that. This has been a first-time process. We are trying to go through it and work with industry to make sure we get to the success, which is a safe plant design, completed and installed and operated in the future.

So --

MEMBER RAY: You used the word "verify down the road," you used that phrase twice. I'm more interested -- I mean it's important to get the design right, and you talked about doing that. We've had a lot of discussion about that. It looks quite comprehensive.

But it seems daunting to imagine how you verify that what is installed in the plant is actually what you intend, that all of this isolation and all of these features actually get implemented.

At what stage, is it now or is it later, that the demonstration of the -- not some prototype, but demonstration of the installed plant systems actually do what we intend for them to do, and don't ever do something different?

MR. JUNG: I agree with that, the challenge of that information design details, the verification activities, they are going to be significant from the staff's perspective. Of course, from the COL applicants and other's perspective when they implement these design acceptance criteria, the amount of documentation they have to develop, the amount of the work they have to do to get to the design, is extremely --

MEMBER RAY: Well, is that something you deal with later, or do you think about it now?

MR. JUNG: No, that's the second part I was getting to.

MEMBER RAY: Okay. Good.

MR. JUNG: So we are working with GEH and also with the industry in terms of the -- when and how much, you know, the staff's involvement has to be there. So maybe our process, PTP 7-14, has a life cycle approach, the guidance related to a specific life cycle stages describe what GEH has provided as an example. Each life cycle stage is from the planning through all the way down to the operational stage.

Each life cycle stage, they will develop a certain -- when they are done with those life cycle stages, that's when the staff is going to be involved to verify that the design information is in compliance with the regulation and the acceptance criteria we have developed.

Not only that, for --

MEMBER APOSTOLAKIS: This is all for the future. Can you tell us what's going on with slide 7? Do you have any conclusions there?

MR. JUNG: Oh, yes, I'll just go over that.

MEMBER APOSTOLAKIS: Okay. Let's do it.

MR. JUNG: Okay.

MEMBER APOSTOLAKIS: Unless there is something very important you want to say before.

MEMBER RAY: Well, George, it may be about the future, it may not. I guess I'm just wondering at what point we hear about how it's going to be shown that the plant actually is --

MEMBER APOSTOLAKIS: And I think that's a good question. But we've spent 10 minutes talking.

MEMBER RAY: We're talking about ITAAC, and that's what I'm asking about.

MR. JUNG: I understand. Let me just complete this page.

MEMBER CORRADINI: If I could just rephrase Harold's question, I think the nub of it is within the ITAAC DAC structure, can you at least give him some assurance that there are certain checkpoints now that you know what you're looking at, so that in the future those checkpoints then are followed through? I think that's --

MEMBER APOSTOLAKIS: I thought the question was different.

MEMBER CORRADINI: Oh, I'm sorry.

(Laughter.)

MEMBER RAY: The majority wants to move on, so --

MEMBER APOSTOLAKIS: I thought you got your answer, which was no answer. All right. Anyway, ask it again.

MEMBER RAY: The tough part of this is

perhaps designing it, but it's equally daunting to figure out how to verify that what you've got in the plant is what you intended, and not something else.

MR. JUNG: The bottom line answer is yes, we do have a plan, our DAC, ITAAC, the GEH has proposed that allows that to happen, and staff is continuously working on that subject to make sure staff has a sufficient time and resources to be able to look at the design details that are sufficient to verify the detailed design is safe and sound.

MEMBER RAY: Well, I'm speaking of a testing program, but go -- move on.

MR. JUNG: Okay. Looking at this slide, I mean it's an oversimplification of what we went through yesterday. The bottom line is staff looked at the -- what GEH has provided in the DCD. We have looked at one of the key items, IEEE-603 compliance issue.

We have looked at the life cycle design process. We just had a discussion. And the set point methodology depends on that. Data communication between the divisions and the safety, it is very important in the digital systems.

We looked at all those areas, and the bottom line is what they have submitted is in a high level of compliance with the regulation and our acceptance criteria.

Yes, the design details have not been implemented, but GEH has proposed the corresponding design acceptance criteria for each of these items, except for set point methodology, looking at the conventional ITAAC.

And we are still working on some of the verbiage here of these DAC they have proposed, but although we believe we are on a successful path to be able to go through the licensing process and make the safety findings based on what they have submitted alone, with the design acceptance criteria, which is the design basis for the future COL applicants to demonstrate that, and we are on a successful path.

The next slide.

The next slide just basically says that we still have 70, plus or minus. We are continuously going through RAIs process. We expect even additional RAIs as needed because, you know, these 70 RAIs actually are substantial work for GEH to make sure their DCD tier 1 and tier 2 documents are consistent and clear, especially on its own design acceptance criteria.

Once we look at that, we will continuously iterate to make sure we get to the clear, you know, safety finding based on that language.

MEMBER CORRADINI: So your two bullets are as you had yesterday, but in the discussion of your two bullets yesterday, a few things popped out, and I want to make sure that the whole committee hears it, so that we're not going to go off in a discussion later, perhaps, in a different way.

One that I heard was it seems minor, but I heard it was major, that you are seeking by current open items and potentially other RAIs a great deal of clarification, so that what I heard yesterday was there's going to be a DCD five-plus, five-point X in terms of tier 1, so there's more clarification as to what the -- in terms of the ITAAC DACs are you going to expect and feel comfortable with relative to the acceptance criteria. That's what I heard yesterday.

MR. JUNG: That's correct, and actually tier 2, also.

MEMBER CORRADINI: Okay.

MR. JUNG: Because tier 2 captures what's in tier 1.

MEMBER CORRADINI: Right. And the second thing I heard yesterday was that there's going to be a cross-reference table which connects essentially --

I'm just only talking about chapter 7 sort of stuff -- but what is in 14.3 relative to DAC, so that there's -- we understand what things in terms of design acceptance criteria and the more details in tier 1 connect to the systems and analysis in tier 2, I heard.

MS. CUBBAGE: IEEE-603 compliance?

MEMBER CORRADINI: Right.

MS. CUBBAGE: Matric?

MR. JUNG: Yes, because as you put it, 603, all individual items of 603 are -- we wanted GEH to spell out as part of their DAC language. That should be cross-referenced in -- for chapter 14.

MEMBER CORRADINI: Okay. And then the third thing I heard yesterday in the discussion of the two bullets was that there were -- and I'm going to get words wrong -- there were functional logic diagrams that were sent to you to help clarify the verbiage in the chapter 7 of the DCD that, although not intimately part of the current review, help you understand and could help us understand the functional logic of the distributed control and instrumentation system that --

MR. JUNG: We'll provide the committee a copy of that.

MEMBER CORRADINI: Okay. Those are the three things I heard yesterday, for the full committee, that I thought at least for some of us gave us a bit more confidence.

CHAIR SHACK: Again, you know, when we go through things like the IEEE compliance, you know, that will be sure we have the independence. Do you think you have enough information to capture all the features that we've heard about today? I mean there are different ways to meet the IEEE-603.

Now they have described a number of features here that seem very attractive, but I'm not sure that they are all captured in the tier 2 information and thus correspondingly in the DAC.

MR. JUNG: Right now, the answer is no, not in -- I would call that sort of sporadic in some areas, because GEH's design stage is depending on areas. The human factors engineering has -- is in progress status right now, for example.

But I think that the issue of, you know, how an applicant will achieve the goal for ICS systems, for example, there's a functional requirement that has been listed in the first system chapter as well as the I&C being supporting area, which goes into 603 and all those requirements.

And if we really want every detail on how one can achieve a certain criteria, 603 is obviously a lot more detailed than GDC, and 603 actually goes into single failure criteria, and then it will go -- our Reg Guide also references the IEEE-379.

So those -- there are a lot of -- the bottom line is that we believe there are a lot of balances and checks in terms of the end goal of achieving what specific things have to be done to demonstrate the compliance or conformance with those criteria.

We believe that from the staff's perspective, if somebody can achieve that, that's what our guidance is, and if you achieve that, we believe it's safe enough.

But do we -- as a regulator, we do not really tell the industry, especially on an evolving technology, you know, how you're going to do that on a design certification stage. We have to picture ourselves 10 to 15 years down the road. The design certification -- it goes way beyond that, so the bottom line is that with the system components that are already set in DCD, and they achieved that going through this process.

MEMBER APOSTOLAKIS: So are you saying

that it's not as simple, straightforward matter to verify compliance with 603? That you have to use some judgment to make sure that what they propose actually meets the functional requirements?

MR. JUNG: There's some cases there's judgments, there's some cases in black and white. You know, the independence separation type of things, the criteria is clear. You just can't -- you know, the nonsafety systems cannot impact the safety function, and we will verify those, and GEH's case, actually those -- you know, GEH, as they explained this morning, they really limit those communications and even call the communication.

CHAIR SHACK: But that's how they're achieving the independence, and I would like to make sure -- I think that's a nifty feature and I like that, and I want to make sure that that is captured or at least it seems to me that that kind of detail of how they're going to meet the high level requirement is part of what I mean by a design.

I mean, every -- you know, a guy could walk in and say I'm going to meet ASME Code IEEE something or other, you know, give me a license approval, you know. They're committing to meet it, it seems to me, at a certain level, and it's -- you know, I don't care if they're using MARK 6 controllers, but I kind of like the notion of what they described in terms of their isolation, and I would like to make sure that those features are captured in the tier 2 description and the corresponding DAC.

And I guess that's the question: Do you feel that -- I mean, you're sounding as though you're hearing some of this for the first time, so obviously it isn't captured in the tier 2 information you're looking at.

Now in the five plus information --

MR. LI: May I make a comment? I think the functional requirements already are specified in the DCDs, and the DCD of the ESBWR is equivalent to the FSAR for the operating plant. So for the NRC regulation review, that's a level that we have.

And the IEEE requirement is a quality of these hardware, software requirement, and it's subject to the ODLE. So you kind of have to go through testing or kind of on-site testing, so it's licensee and GE's responsibility to demonstrate they satisfy all those requirements.

We always have an avenue, and they have to -- they must say to us they are meeting --

CHAIR SHACK: Well, you know, is this

shared memory concept for the internal, external firewall something that is captured as a particular design feature of the ESBWR?

MEMBER RAY: Exactly.

CHAIR SHACK: And, you know, the IEEE says I should keep them isolated. You know, they've picked this particular feature. Now will that be captured as part of the design?

MEMBER CORRADINI: Do you understand where we're going?

MR. LI: I understand. To be licensed, that's -- as long as they satisfy the separation requirement and demonstrate to us.

MEMBER CORRADINI: So what I just heard in your answer is no.

(Laughter.)

MS. CUBBAGE: If we could take a step back, I think what caused some of the confusion, quite frankly, yesterday is GE was not clear in their presentation of what they were presenting was conceptual and what were they presenting that was actually before us for review and approval. And I think we owe you an answer back on what of the details you heard was conceptual and what is going to be fixed, and we'll have to take a step back and take a look at that.

I think what you are hearing from Hulbert is he thinks he satisfied, subject to the resolution of the RAIs, that the level of detail that's captured gives him assurance that the design will meet the regulations and the design will be safe.

But I hear the committee's comments, and we'd like to take a look at that.

MEMBER BLEY: So this will be something for another meeting?

MS. CUBBAGE: Well, we have to come back with all of these chapters at the FSER stage. This is an SER with open items, and we're looking for an interim letter to see if you have any concerns beyond which the staff has identified, and this is a concern that you have.

MEMBER CORRADINI: I think Amy, unless the members disagree, I think she's captured it well in terms of what is conceptual that we heard as examples to give us a warm and fuzzy feeling versus what is going to be captured in the tier 2 and tier 1.

CHAIR SHACK: But Hulbert is sort of giving me a different -- you know, he's satisfied at a different level than I think we're looking at, you know.

MEMBER CORRADINI: Right.

CHAIR SHACK: He's satisfied as long as the IEEE independence is met. The shared memory is a conceptual way to do that at the moment, and he's not willing to nail that down as a feature of the ESBWR system.

MEMBER CORRADINI: Can I say it a different way just back at you? He's saying that he doesn't feel that it needs to be captured in the DCD tier 2-1 structure. Although he's heard it, it gives him a warm and fuzzy, he doesn't need to be there. We somehow feel that -- at least some of us feel that it would be nice to see it there, so we would feel more concrete.

MS. CUBBAGE: And I'm by no stretch of the imagination an I&C expert, but I don't know the extent to which some of these features lock them into a technology, so that could be a consideration why GE has proposed not to specify that. I don't know that for sure, though.

MEMBER BROWN: This is very technology independent, what we're asking for, and what you've asked for, and what others have commented, it's technology independent. I mean the idea that just from the standpoint of voting operation, for instance, they show a line going in between the divisions to show the voting going to two out of the four voting.

CHAIR SHACK: That's a principle.

MEMBER BROWN: That's fine, but does that go -- does that line input software into the other thing, or is it a hard-coded line that goes to a two out of four analog style logic device, which is far more independent and far more isolated than tossing the software into the other software loop, or that loop where they're bringing all the RPS stuff.

Remember, they have all that stuff in that ring bus distribution, and it's rolling around. It's got to be picked up and it's gone to be done something with. That's different than a hard-coded, turn something on, it latches, and away you go, in some two out of -- that's the level of detail that we're looking not. Not the technology that whether they use opto couplers or whether they use some new FEMTO technology that arises here in the next 10 years. I don't care.

What Chip said, you don't care. It's that high level architecture that you're looking for in terms of detail.

CHAIR SHACK: This is an ESBWR DCIS rather than a generic IEEE-603 system.

MEMBER BROWN: I mean, a couple of comments in the DCD said "a function could be." Not "the function is," "a function could be." In other words, a way to do something.

MS. CUBBAGE: I know we're running late on time. I don't know whether you want to give an GE an opportunity to try to explain briefly what they presented yesterday was conceptual versus what they are committing to in the DCD.

MEMBER CORRADINI: I would say given the time -- and I don't want Dr. Powers to remember that I'm too late -- I would take it as an action item that -- what you said, which is what conceptual versus what is the action DCD in terms of design features, something that we will expect to hear from the staff after you have discussed with GEH.

MEMBER STETKAR: I don't know if I should --

(Laughter.)

I want to follow up just a little bit on what Bill said. And to give you specific examples of this. And I think -- I don't think our role as ACRS is to get into microfine structure detail of the design, which is unfortunately a natural phenomenon every time you talk about I&C. I&C people love to get into fine structure.

I think our purpose -- this is my own personal opinion -- is to step back and look at the integrated design of I&C within the context of the plant, from a safety perspective, from a reasonableness of design perspective of whatever you want to call it. And that requires something more than what's available in the DCD, and it certainly does not require the final detailed design.

You know, in all relay-driven technology, I don't care whether I have a six-contact Westinghouse relay or a four-contact GE relay, if I only need four contacts, it doesn't make any difference. I don't care.

Now a couple of examples of what do I mean by this. Briefly.

For example, in the current design, an ESBWR uses power at the -- whatever they call them, power generation buses as the measure of loss of feedwater flow, because it is assumed that that is the only way you can lose feedwater.

Question: Why don't they use feedwater flow as the input, because there are many other ways of using feedwater.

Now that's a medium level of detail. It

isn't designed -- it isn't final hardware, software specific, it is an input functional signal. You can say it's high level, medium level, I don't care what it is, why are they using loss of power at a bus as a surrogate for loss of feedwater flow?

So that's one level of information.

Another level of information is there are probably a dozen different ways that you can meet single failure or even double criteria. There are some that are better than others in terms of distribution of signals among pieces of equipment.

A little bit of that came up when we talked about the DCD manual initiation that touching one division actuates less than all of the equipment.

That is, again, an intermediate level of detail that is not currently in the DCD. It doesn't depend on software or hardware architecture. It's a design specification.

If you can look at that level of detail and say, yes, they seem to have accounted for all the different ways you can lose feedwater because of the signals that they use as an input. Yes, their design specification for how they distributed the signals among different pieces of equipment satisfies system interactions, both within the I&C system, and within different systems in the plant.

I think that we would feel a lot more comfortable about saying that, yes, indeed, the design is robust and that those criteria then can be verified when the thing is finally built. Because, you know, how do you do the logic is a matter of construction rather -- it's how it's implemented is the final construction. It's not affected by the original decisions.

That's enough.

MEMBER BROWN: But you've got information in the DCD that allows them to --

MEMBER STETKAR: That's right.

MEMBER BROWN: -- that status profile and what they test to verify. And that's -- the point is that's not their --

MEMBER STETKAR: Right.

MEMBER CORRADINI: So I was going to ask for member comments, but I think I got them, anyway.

Are there any questions of the staff at this point? Beyond what we have just gone through?

Therefore, Mr. Chairman, I want to thank the staff and GEH for their summary presentation, and I remind everybody we have the summary presentation at the last meeting in chapter 14, and together we will come up with something we hope today.

CHAIR SHACK: We will take a break for 10 minutes, and I would like to remind those of you who haven't seen Sherry yet that she's looking forward to meeting you.

(Recess.)

CHAIR SHACK: Let's come back to the session.

Our next topic is the early site permit application and the final SER for the Vogtle Nuclear Plant, and Dana will be leading us through this matter.

MEMBER POWERS: Thank you. Let me begin by noting that in this matter, Said has a conflict of interest, and though he is welcome to comment for purpose of qualification and purpose of opinion, he is not to be listened to.

(Laughter.)

We are here to finalize our review of the Vogtle early site permit. This is in fact the fourth early site permit that the committee has reviewed. Some of the members, I note, have never had the opportunity to participate in an early site permit review, so this will be exciting.

The application has some unusual features to it, that instead of calling it a plant parameter envelope or site characterization, it calls out a specific plant, the AP1000.

It also is unusual in the sense that it calls out a complete integrated emergency plan rather than just the major features of an emergency plan.

And in this particular application, there is also an unusual feature that they have conjoined with the early site permit a limited work authorization under the provisions of a relatively new rule.

So it has some unusual features to it.

The modifications to the site that have been proposed are dramatic. I'll let the speakers discuss that, but not unusual in the sense that they are analogous to modifications that have been made to adjacent units.

The subcommittee met yesterday to discuss this thoroughly. Quite frankly, we have been over the major portions of the early site permit request both in subcommittee and in full committee, but I have asked the speakers to review some of that material, but there are so many members who have not seen it before, and in many respects you can just trust that the committee -- subcommittee plunged into the details of closing out various orphan items, and we may be purely summary in our presentation of those.

The dominant list at this site, as with most -- not all, but most -- early site permits is a seismic source term posed by the Charleston seismic zone, and we did discuss some of that.

Like all seismic zones on the East Coast in the central United States, they are a bit more mysterious than those in California. But don't ask for the details that you would in the San Andreas Fault.

MEMBER RYAN: All righty, then.

(Laughter.)

MEMBER POWERS: The subcommittee has formulated a draft position with respect to this early site permit. Our draft position is that the Vogtle site is suitable for the location of a light-water -- two new light-water reactors.

We note, however, the projected ground motion seismic response spectrum is not bounded by a good design spectrum for the AP1000 reactor, and that would be something that would have to be resolved in the future if that's the plant they want to locate there.

What we are proposing is our draft

presentation with the emergency plan is acceptable, and that a limited work authorization is also acceptable, and analyses have shown that even with an AP1000 on this site, it would not be subject to overturn or sliding as a result of the proposed work.

So bear those positions in mind as the speakers go through that.

I have asked the applicant and the staff to somewhat coordinate their presentations because they are trying to summarize in a relatively limited time material that it took us a full day to go through, as well as providing background for members that have not seen this before.

We are going to try to accomplish quite a bit here in a limited period of time. So with that, I will turn to Mr. Davis to give us some background on the site.

MR. DAVIS: I do apologize, we're having somebody retrieve our electronic files, but if everybody has a handout similar to this, this is what we're going to be going through, and I'll just walk you through it. I'll try and say the number of the slide.

I am Jim Davis. I am the ESP project manager for Southern Nuclear. I am responsible for the development of the ESP application.

If you will flip the page to part two.

As Dr. Powers mentioned, we are trying to divide up the presentation material since we have both covered a lot of the same stuff with the NRC and the applicant yesterday, so I'll try and give an overview of what the content of the application and the type of things that are in there, and later I'll turn it over to Christian to talk more about the technical details of what they evaluated.

So basically I am going to cover a short introduction, I'll give you a feel for our schedule of activities, when we are going to begin construction, some of the preconstruction activities, a quick overview of the early site permit and its content, and an overview of our limited work authorization request.

Southern Nuclear has submitted an early site permit in accordance with 10 CFR 52, subpart (a) for early site permits, and also as Dr. Powers mentioned, we have requested a limited work authorization under the new rule for 10 CFR 50.10.

Basically the ESP grants -- if you'll go to page 3 -- the ESP grants approval of a site for one or more nuclear power facilities separate from the filing of an application for a construction permit, or a combined license.

I think my slides just showed up.

MR. WIDMAYER: As long as they weren't x-rayed or anything.

(Laughter.)

MR. DAVIS: No, no x-rays. The kids' pictures are on there, too. We'll try not to load those up.

(Pause.)

On slide 6 on page 3, the Vogtle application is the fourth ESP application that has been pursued under the new Part 52, and basically Dr. Powers mentioned some of the differences.

The first three applicants had a plant parameter envelope approach where they wanted to qualify their site for multiple designs for a nuclear power plants.

The difference for ours is we took a look at the first three applicants. We tried to do our lessons learned based on what issues cropped up for them. The environment report specifically. It's hard to draw a conclusion based on a parameter envelope, so we decided to go with a specific technology, and the technology that Southern Nuclear selected was the Westinghouse AP1000.

Basically what that allowed us to do was have specific conceptual design developed for our site so that we could evaluate our site to a specific -- particular design, which was the AP1000, and that allowed us to pursue things more in depth and more in detail.

The other thing that Dr. Powers mentioned is that we chose, instead of a major -- for our emergency planning, we chose to do the detailed complete and integrated emergency plan, and the first three applicants did not pursue an LWA, which Vogtle has done.

So we have a lot of differences from the first three. We tried to learn from how they did it and tried and come up with a better, more complete application that would allow us to achieve more finality.

Basically, just if you will flip to page 4, slide 7, it's just an overview of our schedule. If you'll take a look at -- we put about 19 months into developing the ESP application, which we submitted in August of '06, and since August '06 until today, the NRC has been reviewing it and going through our application. We've been working with them.

Basically we're hoping for the ESP and LWA in the fall of 2009, which would support our construction schedule. And I'll go into a little more detail on the schedule as we proceed through the slides.

The Vogtle site is an existing two-unit nuclear facility. It's a 3,169-acre site. It's in Burke County, Georgia. It's located on the Savannah River, directly across the river from the Savannah River site, the DOE site, and it's about 26 miles southeast of Augusta, Georgia, and approximately 150 river miles from the port of Savannah.

Just to give you a little better perspective where that is, you can see Augusta, Georgia above the site, and if you look at the little legend on the right-hand side, you'll see where it is on the border between Georgia and South Carolina, approximately where it is.

Basically the early site permit was five parts. The first part is the introduction which identifies the owners and who is applying for the application, a few administrative details.

Part two is the safety analysis report. This is where most of the safety work is done -- evaluation is done along with part five, which is the emergency plan.

This gives all the details of our site, how we analyze the site, and I'll go into a little more detail about what's in that part.

Part three was a complete environmental report where we evaluated both the impacts to the environment during construction and operation, so that we had a complete scope of the environmental impacts due to the addition of two additional units at the Vogtle site.

And part four is a redress plan. That's what is necessary if you want to pursue an LWA. It's kind of remediation. If you for some reason decide not to pursue construction of the site, it's what you would do to bring the site to an acceptable condition.

And then part five, of course, which I mentioned, was the complete and integrated emergency plan.

Getting into a little more detail about what's actually in the early site permit. Basically we followed the same format as an FSAR for an operating unit. And the sections that are included in the ESP are those sections necessary to support evaluation of the site and support our LWA.

So as you can see here, we have chapter 1, which is general introductions. Chapter 2 and 3 probably contain the most information. It's the largest section because it's actually evaluating our site characteristics, are they acceptable for the design we propose to put on the site.

Chapter 3, we have a little more detail than normal in an ESP because we have some foundation information that's necessary to support our LWA activities that we have requested, so there's a little more there than in a normal ESP.

We also have rad waste evaluations on liquid and gaseous effluents, and chapter 13 is a little more than normal for an ESP as well because when you request an LWA, you also have to have the programs in place to support those activities, specifically your fitness-for-duty program.

So there are certain elements that are a little more than a normal ESP because of the LWA.

Also chapter 15 is accident analysis, and 17 is our quality assurance program.

Here is an overview, a sunlight picture of the site, with a little artist's rendering that shows our new units. The existing units, 1 and 2, are on the right.

To the west of those units is our proposed unit 3 and 4. We have an existing intake structure --

there we go.

(Pause.)

The first structure is the existing one. I can barely -- I don't know if you -- this is existing structure, the existing units 1 and 2, where we are putting in a new intake structure for 3 and 4.

And this is the new units 3 and 4, which are going to be to the west of the existing units.

MEMBER BANERJEE: What size are the existing units?

MR. DAVIS: They're about, what is it, 1240?

MEMBER SIEBER: About. Four-loop.

MR. DAVIS: Yes, four-loop PWR Westinghouse units.

MEMBER STETKAR: Are the existing units and the new units going to share the same switchyard, 500 kV switchyard?

MR. DAVIS: We're going to expand the switchyard. We're going to have a new 500 kV line coming into the switchyard. They will be physically connected, so even though this looks like it's separate, they really are connected, but they're kind of stretched out. Just, you know, from logistics.

Actually the current plan is that

eventually we will swap out some of the transmission lines that are going to 1 and 2, one of those lines will go to 3 and 4.

MEMBER BANERJEE: That's okay.

MEMBER SIEBER: In addition, you have two lines, 230 kV lines?

MR. DAVIS: We have a 500 kV line, and then we have a set of 230 kV lines coming in.

MEMBER SIEBER: And that kV lines goes to --

MR. DAVIS: The current one is the Macintosh. It goes south down towards Savannah. And we're going to add a new -- we have a 500 kV line that goes to Share, so I think there's two 500 kVs, and some 230 kVs. We're adding an additional 500 kV which is going up towards the Augusta area, west of Augusta, to handle the transmission.

MEMBER SIEBER: West of Fort Gordon?

MR. DAVIS: West of Fort Gordon, yes.

This next slide just illustrates the fact that we are a little bit different from a lot of the other COL applicants in that we are a deep soil site. Most of the other applicants are rock sites, which was analyzed specifically in the DCD. It made their seismic evaluation a little easier than ours. Even the soil sites that were evaluated by the DCD don't go near as deep as ours do.

We're in the coastal plains, and we have over 1,000 feet of soil above our bedrock, so that kind of complicated things when we did our seismic evaluation. But this is just a slide to illustrate that. It's not to scale.

Basically we had a very rigorous review by the NRC, and they did a very good job. We tried to work the best we could with them. We didn't always agreement on the method or level of detail --

(Laughter.)

MEMBER POWERS: What a shock.

(Laughter.)

MR. DAVIS: But we both eventually arrived at the same conclusion, that it was a good site to build a plant on.

This just gives you kind of a feel for the number of issues we had to deal with. We had 189 RAIs prior to receiving our SER with open items, and these are the numbers associated with the different subject areas.

If you'll notice in tier 5, geology got the most RAIs.

And then we received our SER with open

items. You again can see that geology was our favorite subject, so we spent a lot of time on that. We've worked with NRC to reach resolution on these open items.

MEMBER BANERJEE: Did you have to do a lot more work compared to what you'd done for the other two units?

MEMBER SIEBER: Yes.

MR. DAVIS: Yes. Much more.

Part of what the timing on our LWA, initially we -- and I'll get into this in a little more detail -- we requested an expanded LWA for the safety-related work right about the time they were issuing the SER with open items, and as a result we got 26 more RAIs specifically with our expanded scope LWA.

Basically they covered site investigation information, enduring properties of the subsurface materials, and the backfill requirements, the design of the engineering field we wanted to put in as part of that LWA.

MEMBER BANERJEE: So between the time of these two submittals, we learned a lot about geology?

MR. DAVIS: Yes.

(Laughter.)

MR. DAVIS: I have a lot of experts that work for me, and I've been on a lot of phone calls, and I would say that I know a lot more than I ever did, or maybe even wanted to, when we first started to.

MEMBER SIEBER: You learned more about geology and you learned more about your site.

MR. DAVIS: That's true. We know a lot about the Vogtle site. In fact, I would be surprised if anybody knows as much as we do about their site. It's been very thoroughly investigated.

MEMBER POWERS: I don't know, the Clinton folks probably will -- they'd be willing to match page for page, I suspect.

(Laughter.)

MR. DAVIS: Now I'm going to cover a little bit about the LWA and preconstruction just to give you a feel for what we asked for, and about what our schedule is right now.

Basically if you look at our initial submittal in August of 2006, we requested an LWA-1 under the old rule. An LWA-1 basically is like preparation for construction -- building the roads, putting the infrastructure in, facilities, and warehouses to support construction.

Then in August of 2007, we requested the LWA-2, which expanded the request to a limited scope safety-related work activities.

Then once the new rule came out, we converted all that to the new LWA, and under the new rule basically you don't have to have an LWA for preconstruction activities, but you do have to have an LWA for safety related.

So we converted our LWA-1 and LWA-2 to the new rule.

Basically I'll just kind of hit some highlights on preconstruction activities.

This is actually a list from -- directly from the regulation. It says what kind of things are not defined as construction, and as you can see, it's site exploration, kind of like the borings and stuff that we do when we're trying to qualify a site. A lot of the preparation for is developing the construction, infrastructure including grading and drainage and other things.

And just to point out, the excavation is something that is not considered construction. We can dig the big hole. We can't put anything in it, but we can dig the big hole.

Basically we can do parking lots, railroad spurs, transmission lines. You know, we talked a little bit about you can do those without an LWA.

Also one thing I will point out, the last bullet, some of this new concept for the passive plants is dependent on modular construction. So as long as a module is being built, there's a fabrication shop off site, or assembly into a big module on site, those things aren't considered construction. We plan to have some laydown areas for that activity to go on.

Basically what we have asked for --

CHAIR SHACK: Say that one again.

MR. DAVIS: If we have modules, okay, which are plant pre-fabbed assemblies, if we had a contract with a vendor to build that for us, okay, that's not covered under the LWA or it's not considered construction until it's placed in its final location.

So some of our vendors, subsuppliers, we're going to allow space on site to do some final assembly to stage the components for the construction.

Now I won't say that that timing actually brings them in before we get our LWA or our COL, but there might be some staged materials on site, and that's not considered construction until you put it in place and start connecting everything.

MEMBER POWERS: It's all totally reasonable.

MR. DAVIS: The things that we have asked for, because we had such a large excavation -- and I've got several slides to kind of illustrate that for you, and the time period it's going to take us to dig the hole and then fill the hole back up, that was the reason, the primary reason, that we pursued the LWA. And so we asked for engineered backfill under our LWA, retaining walls -- and I'll give you an illustration of this -- mechanically stabilized earth wall, which we're going to do, leaning concrete backfill for around any things that are hard to use soil to backfill around.

We're going to put in some mud mats underneath the nuclear island. We're also going to install some waterproofing on those mud mats and some of the walls which will actually be forms for the nuclear island walls.

Along with that, I mentioned earlier you have to have certain programs to do safety-related work, including our fitness-for-duty program, our QA program, and also our problem identification and resolution or corrective action program.

Just to give you a feel for what our near-term schedule is and some of our key milestones at the beginning of construction, for us a key milestone is our PSC approval. That allows us to actually spend money. You know, even though we can't do any safety-related work --

CHAIR SHACK: You get reimbursed.

MR. DAVIS: That's right.

(Laughter.)

MR. DAVIS: So that's a key milestone for us to actually begin spending money.

Also the ESP approval, with the other day, we're anticipating that we'll get that in the fall of '09, and then actually our COL application, which we hope to get in the fall of 2011.

Basically we are already starting some of our preconstruction activities. We are doing demolition of old structures and slabs that are in the footprint of the new units. So we've already got that started, but when we really want to spend big money is the excavation of the hole, which we are going to remove about 3.6 million cubic yards of material. So we've got a very large excavation. It's going to take us about six months to dig the hole.

Once the hole is finished, we are going to do some geological mapping because it's going to expose all those layers of material, and at that point in time the NRC is going to be invited to come down and evaluate the site and take a look at that aspect of our program.

And late in 2009, we will begin the backfill, backfilling the hole, proceed from our bearing layer, and I've got a couple of slides to illustrate this, of about 50 feet, so we have to go down approximately 80 to 90 feet, and then we come back up with engineered fill to about 50 feet, and then is when we start putting in the mud slabs for the nuclear island.

We will start an MSE wall, which will actually be kind of the form, the outside dimensions of the slab, and continue the MSE wall up the grade. We'll pour a mud mat inside of these walls for -- which will go underneath the nuclear island. We'll install waterproofing on that mud slab and beginning at the walls, and we'll do a work surface mud mat on top of the waterproofing to protect that from the rebar installation.

Again, this is just a layout. You've seen this slide again. I'm just going to show you the following slide.

Here is the excavation, just an

illustration of how big the holes are in relationship to the -- next. Basically here is another slide that shows. And the reason we had to excavate our site was because we had limestone above our bearing layer and the upper sands were were subject to liquefaction, and I think Christian has got a real nice slide on that later to illustrate what that is.

But during a seismic event, the upper sands have the potential of liquefying, so you don't want to build a unit on it. That's why we're taking it all out.

This shows you the extent of the hole, and we've got some cross-sections on it. Basically how we established what the bottom of the hole was going to be is we took a zone of influence from all the buildings, and we took the outside corners of the buildings, and we drew a 45-degree angle coming down to the bearing layer, and that's how we established the outside dimensions of the wall.

And then we're going to bring the fill back up -- this kind of illustrates the 50 feet that comes up from the wall, and then we start the mud mat, and the MSE walls will come up the side, and we're going to bring it up to the grade.

MEMBER BANERJEE: So where do you get all the material that you're going to use to backfill this?

MR. DAVIS: We have -- out of that excavation we hope to save -- I think it's about 40 percent is what we estimate, and we have other borrow areas that we identified during the ESP base where we have more material that's acceptable, and as it comes out of the holes, we will have like soils labs to test the material and we'll segregate it. The stuff that we can use, we'll put in a borrow pile. The stuff we can't use we'll put to a spoils pile.

MEMBER BANERJEE: Well, is the other 60 percent coming from off your site?

MR. DAVIS: No. All -- we plan on having to retrieve all material, fill material, from our own site. We've got a 3,169-acre site, so we've identified other sources of borrow material on our land.

We have --

CHAIR SHACK: Now when you built the previous units, you took the foundation all the way down to the blue marl, and you're not doing that now?

MR. DAVIS: Well, I would like to thank you for that.

CHAIR SHACK: You're my straight man.

MR. DAVIS: That's right. Yesterday we did mention that some of our facilities are on the marl. The aux building, the NECW towers are on the marl, but the actual -- like the container building, I think it has like about 26 feet of fill material, just like the reactors here will have. The ones we have now are 50 feet, but they actually are on the fill, certain portions of the plant are on the fill, including the reactor building for unit 1 and 2.

MEMBER ARMIJO: Now one thing I missed, I'm just looking at your previous drawing, the turbine building is not on the same -- will not have this mud mat, deep rock, so its foundation will be at a higher level.

MR. DAVIS: Yes.

MEMBER ARMIJO: Okay.

MR. DAVIS: The nuclear island is the deepest structure, and that's the first -- you know, the first level we get to that has a structure on it is 50 feet, it's the nuclear island.

Most of the rest of them are much nearer the surface.

As part of our ESP and COLA process, we actually did do a test-fill program for the MSE wall. We wanted to demonstrate that the small equipment in our backfill material would be acceptable. We brought in some industry, you know, experts to help us with this, plus some of our soils people to test our material as we put it in.

Basically we were trying to figure out if our methodology would work the way we expected it to.

And it's not new technology, MSE walls. You see them everywhere. Here's one near -- on the Atlantic Expressway near the airport, so it's nothing unique or special about them. We're just taking them and applying them to our site.

Here's just an example of the waterproof membrane that we're going to be using. It's an elastomeric spray-on product that goes on in two layers, and once we get the MSE wall started and the mud mats poured, we'll put in a layer of this material, and it will actually go up the side of the walls as the MSE walls come to the surface.

Basically when we get done, this is -- this is what we're trying to achieve to prepare ourselves for construction. We're going to have a big swimming pool out there. The backfill is going to be up to grade, and we're going to be ready to roll once we got our COL.

And because it takes, you know, a year and a half to do the backfill, we're trying to get a jump-start on that so we'll be ready for construction.

MEMBER SIEBER: Did you ever get the question resolved about the issue of the grounding mat?

MR. DAVIS: We did follow up on that last night, and there will be a grounding mat put in late.

MEMBER SIEBER: Under the membrane?

MR. DAVIS: It will penetrate the membrane at some point, okay, but --

MEMBER SIEBER: It's probably at multiple points.

MR. DAVIS: Right. But when we contacted the Westinghouse people about what method they use, they gave us like three options, so I don't know we have settled on a particular one, but each one of them did penetrate.

MEMBER SIEBER: There is a way to do it and keep it waterproofed.

MR. DAVIS: Right. Right. This product was qualified for penetration.

MEMBER STETKAR: Just so people realize, there has to be penetration.

MR. DAVIS: Right. And one thing we pointed out yesterday, this is part of the certified design to have the water proofing in there for in-leakage of water. It's not safety related, but for the Vogtle site we're putting it in, but our normal groundwater elevation is about well below our -- the nuclear baseline.

MEMBER BANERJEE: So you said this is a spray-on. Is it a polymeric system or what does it do? You spray it on and it sort of evaporates across all this stuff?

MR. DAVIS: Yes.

MEMBER POWERS: It's rubber.

MEMBER SIEBER: It's thick, though. Six inches, maybe.

MR. DAVIS: No, it's much thinner than that. And that basically gave -- are there any questions on that? If not, I'll turn it over to Christian.

MEMBER POWERS: Now I'll ask the staff to go into a little more of the detailed evaluations that they did. Needless to say, as Jim pointed out, this is a lengthy application. It required quite a team of people to review because it covers a diversity of things.

One of the things to recognize here is that the seismicity of the area has been studied a lot in the past because of the proximity of Savannah River, and it was studied even more in connection with this early site permit.

(Pause.)

MR. ARAGUAS: All right, my name is Christian Araguas. I'm the project manager for the safety review for the Vogtle ESP application.

As Dana pointed out, we have really truncated the slides so if you guys have any questions, feel free to stop us and ask. It's a lot of material to get through, so trying to do it in a 45-minute timeframe is going to be tough.

And having said, what we are going to cover, as Jim pointed out, is two aspects of this. One is the review of the ESP application and closure of the open items, and the other aspect is the staff's review of its first LWA request under the new rule, and then we will address any questions.

So just really quickly, we wanted to cover the agenda for the next hour. I'll go through some of the remaining milestones for the project, and then at that point I will turn it over to the technical staff to go through how we closed out some of the open items, and then at that point we'll summarize the review of the SER and we'll move on to the LWA discussion, and we'll go through some of the key areas that were reviewed and exactly what it is they are planning to do under their LWA.

So as I mentioned, this slide just really quickly is here to demonstrate what we have left. We're expecting a final letter from the ACRS sometime in the December-January timeframe.

Following that, we will issue the final SER in February 5th of 2009, and then the Atomic Safety & Licensing Board has set its hearing schedule to start March 23rd, 2009.

Following that, we would expect a decision from the ASLB in the I think it's July timeframe, and then depending on what the commission wants to do, if they decide to weigh in or not, we would expect a decision on the issuance of the ESP in the summer or fall timeframe.

Okay. So this slide, I just wanted to show similar to what Jim has already shown. These are the review areas.

On the left-hand side, you will see the areas that we focused on that are strictly for the ESP. On the right-hand side you will see the areas that were looked at for the LWA.

Where we have areas that were highlighted in bold, the areas where we had open items, we don't plan to cover all the open items. As you can see, there were 40 of them, so it's a lot to get through. We will focus on the major issues that we thought were pertinent for today's meeting.

The other key aspect of the slide that I wanted to point out, and it tends to be a little bit confusing because we have two actions going forward, is what exactly you're approving under an ESP, what exactly you're getting under an LWA with respect to the design.

So for an ESP, unlike the previous three applicants, Southern has requested site suitability review done for and has referenced the AP1000 certified design.

When you approve an ESP, it is not allowing approval for that design at that spot. You're just approving the characteristics that were looked at as part of that ESP, such that at the COL stage you're do a comparison to verify that that design will fit on that site.

The LWA, it's a little bit more tricky. With the LWA you're actually pulling out portions of the COL, and that now you have to rely on portions of the design, but only those that are necessary for approval of those specific activities that were requested under the LWA.

So I wanted to highlight that going forward so that way with respect to the LWA, it's only those areas, as Dana pointed out, with respect to sliding and overturning that we knew to look at design aspects.

With that, I will start with our hydrologist.

MR. AHN: My name is Hosung Ahn, hydrologist with the NRC.

I will start with a brief introduction of what we did under SER section 2.4, and what are the major findings of the section. Then I will describe the open item and how we resolved that open item.

Section 2.4 consists of over 14 subsections, and half of them are telling me the flooding issues, maybe it's by either precipitation, the in-break, or hurricane and the tsunamis.

So we analyzed each and every flooding event, and determined what are the impacts of the flooding.

In flooding there are a lot of different parameters, like what is the maximum flooding level, or what is the step in the dynamic forces of the flooding, but one of the critical flooding parameters is the maximum flooding level. And you use a lot of moderating technique to estimate the flooding scenario, and they identified the dam break is the most critical bounding flooding event.

Especially for a dam break, on this side upstream there are about 14 major dams, and they picked the two most high volume reservoirs, and used the cascading dam failure to estimate the flooding level. So that's the consolidated approaches.

A step used for similar modeling approach, even more conservative parameter set.

Therefore, we concluded that the site is safe from any flooding event.

The maximum flooding level they estimate is about 178 feet from the river, and the site grade is about 220 feet, so they have a lot of margin on the flooding, so that's the basis of the safety on flooding.

And we also analyzed the low-flow impact as there was the ice condition, on-site groundwater use for the safety-related water supply, and we identified that we found that for those safety-related water supply are not impacted over this hydrologic hazard.

The last two things dealing with groundwater flow and the radionuclide containment and transport.

MEMBER STETKAR: How far from a hurricane zone is this site?

MR. AHN: It is about 150 miles from the coastal line, and the site is located about 120, so a hurricane may not affect this site. Flooding is not affected on the site.

MEMBER APOSTOLAKIS: So what contaminants is the groundwater transporting?

MR. AHN: To analyze the contamination, we postulated one of the rad waste pipe failure scenario and analyzed the contamination on the receptor area.

The lower left corner features basically show the water table contaminant for the official output, and the bottom layer is bounded by the blue marl, and the aquifer is subject to the radionuclide transport contamination.

So we at the beginning we listed the aquifer system, and the depth groundwater region may be quite sensitive to the change.

MEMBER BANERJEE: Is that picture on top of a real flood in that region?

MR. ARAGUAS: No.

MR. AHN: Not on the site.

(Laughter.)

MEMBER POWERS: I explained to the speakers that there are many new members that might not be familiar with everything.

(Laughter.)

And obviously that's one of the things that we're concerned about.

MR. AHN: These pictures show the --

(Pause.)

-- units, and that area is bounded by the Savannah River and then we have a small creek on that side, and it's also bounded by a small creek on this other side.

There is a groundwater transport pathway, major pathway. However, the extensive excavation and the facility installation, that groundwater region might change. So at the beginning we pushed the -- how that groundwater region may change.

MEMBER RYAN: And in light of that explanation, there was a change in groundwater pattern?

MR. AHN: At the beginning.

MEMBER RYAN: I'm sorry?

MR. AHN: That's what we assume at the

beginning. That's why we asked through the RAI and open items, put your analysis in detail so that how the groundwater changes and how the pathway may change, and what are the alternative potential pathways. These pathways going to the north, but how about to the sites or to west. So we asked the applicant to analyze, to do that. So I'll explain that a little further.

MEMBER RYAN: But that's a work in progress, that's an open item?

MR. AHN: Yes.

MEMBER RYAN: Okay. Thank you. That's fine.

MEMBER POWERS: This is a closed item.

MR. AHN: I'm sorry?

MEMBER POWERS: This is a closed item.

MR. AHN: It was an open item, but we closed that.

MEMBER RYAN: Okay. I just wanted to understand. Thank you.

MR. ARAGUAS: Your question was did the model assume that the back wall had been placed down; is that right?

MEMBER RYAN: My real question is what is the impact of a new island on the groundwater flow.

Have you modeled what that will look like when it's all done?

MR. ARAGUAS: Right.

MEMBER RYAN: That ultimately determines the flows of the pathways, monitoring programs, all that other stuff.

MR. AHN: Let's explain the open item.

MEMBER RYAN: Okay.

MR. AHN: We discussed that in detail yesterday, but I would just briefly introduce that open item, and how we resolved that.

The first open item, 2.4.81, we had the issues that whether the section 2.4.8 is the safety of the canal and reservoir, so that they can provide a safety-related water even during the hazard condition.

So we asked that whether they used the canal or reservoir as a safety-related water supply. And the second question is in case the applicant proposed two water tanks for safety-related water supply, when they're initially fitting the tank, or when they make up water for the tank, is that safety related or not. So that was our initial concern.

So we issued the RAI, and the applicant responds that they are not going to use the river and canal as a safety-related facility, and even initial filling of makeup water is not a safety-related activity.

So -- and they said that for the initial filling makeup, they will use the true groundwater wells, and the water they will extract from the aquifer. So on section 2.4.12, that's the groundwater section, we reviewed based on the hydrologic, whether that aquifer provided enough supplemental water to provide water to the tank. And we found that they have a sufficient -- the aquifer itself has a sufficient capacity. There are aquifers below that aquifer. So we closed that open item No. 1.

The second and the third and the fourth item are related to the groundwater models. In general, from both sides, the hydrologic pathway is very simple, so we may not need a groundwater model, but in this case groundwater could be very sensitive based on the excavation and the facility installation.

That's why we pushed this issue very hard. So open item No. 4 is after the preconstruction, how high the natural groundwater level will change. The applicant estimated about natural groundwater level really 165. That estimation is based on historical data, but historical data doesn't mean anything for future conditions.

So we asked them the groundwater, natural groundwater will change. The applicant, in their response, they provided additional hydro-geologic data based on the COL filing and LWA drilling, and they provided --

MEMBER RYAN: Just before we get to this slide that's behind you on the screen, just so I understand it, the water table elevation is 165 in the cell, and the grade is 220. That's between 165 and 220 in terms of water. Is it an unsaturated zone?

MR. AHN: It's an unsaturated zone.

MEMBER RYAN: How can that be? This is Georgia. The saturated zone is very close to the surface in Georgia and South Carolina.

MR. AHN: On this slide --

MEMBER RYAN: This is just the explanation.

MR. AHN: No, no, that's the current condition, 165.

MEMBER RYAN: The unsaturated zone is 50 feet thick?

MR. AHN: Yes. It's very low.

MEMBER RYAN: Okay.

MR. AHN: And because of the extensive excavation, the rate might increase, so therefore can we make up, that was our initial question. So we asked them to elaborate your estimation, and they estimated that the groundwater level, to validate that value, and then they demonstrate that that area is quite ready for -- at the construction. And the step used more conservation more approaches to estimate that, and we found that there are enough margin.

MEMBER BANERJEE: Are these models pretty well established?

MR. AHN: That was very well effective. It was well documented, and they did the calibration, and it showed it is quite acceptable. So that's why we accepted their model.

We conformed that measurement groundwater level and we closed this open item No. 2.

Open item No. 3. This is quite similar to the previous open item, but it is related to the ultimate conceptual model considering variability and uncertainty on the hydro-geologic parameter. And as I mentioned before, they provided additional data, and the groundwater level, and they confirmed the postulated pathway. And on the part of that step using more conservative hydro-geologic parameter to identify pathway to either the eastern side or the western side, and in an extreme case, the pathway for that is not highly plausible. And at the end of it, we did containment consequence analysis and found that the pathway is the most critical pathway. That's what we found. So we closed that open time, too.

The last open item, we considered the location of the receptor area and the different source location, and made some combinations of a different pathway, alternate cable crossover pathway, and analyzed what is the most significant contamination pathway and what are the consequences, and then we identified that both the pathway to the middle of the pond is again the most critical, but this other pathway Appendix A compliance. So that means the site meet the external release of contamination criteria.

So that is the close of my presentation.

MEMBER APOSTOLAKIS: What is a chelating agent?

MR. AHN: Oh, chelating.

MEMBER CORRADINI: How do you pronounce?

MR. AHN: Chelating.

MR. ARAGUAS: Chelating agent.

MEMBER POWERS: Citric acid, EDTA, nitrilotriacetic acid.

MEMBER BANERJEE: I don't understand what the issue there is. Why is there no chelating agent will be --

MEMBER RYAN: Chelating agents tend to make the radionuclides more mobile, as opposed to ionic or, you know, nonreactive with other things.

MR. AHN: So when we do the consequence analysis, the applicant did his own, without the chelating. That means there's no contamination, there's no chelating agent appeared on there. But when they used the chelating agent and that is mixed with rad waste material, transport will be faster.

MEMBER CORRADINI: What systems in the plant use such agents?

MR. AHN: What systems?

MEMBER POWERS: Any time you do a cleaning.

MEMBER CORRADINI: Okay. Like a steam generator cleaning or --

MEMBER POWERS: Yes, something like that.

MEMBER CORRADINI: Okay.

MR. AHN: So there are two issues, whether that chelating agent will mix with the rad waste material and if that is mixed with rad waste material, we may need to do the analysis again with considering that chelating agent.

MEMBER BANERJEE: So you can separate out the chelating agents if you use it as cleaning from the rad waste? Is that possible?

MEMBER POWERS: You treat it with a little peroxide mix on it.

MEMBER BANERJEE: Okay. So you change the chemical structure.

MEMBER RYAN: Right.

MR. AHN: I think they need some operating the plant, but our concern is whether that chelating agent is mixed with the rad waste or not.

MEMBER BANERJEE: The effluent stream.

MR. AHN: Yes, effluent. And what are the impact of that.

MEMBER BANERJEE: Go ahead. Thank you. Very well.

MR. AHN: So for that, NRC said we may need more detailed design on the condition and the structure in there. That's why we put that as a COL action item.

MR. STIREWALT: Good morning. I am Gary Stirewalt, and I would like to talk just briefly really about the sorts of things that we reviewed in section 2.5.

MR. ARAGUAS: Hey, Gary, can you back up a slide? There you go.

MR. STIREWALT: But again, just to sort of set the scene for the sorts of things that we really reviewed in section 2.5 now, basically -- and Mr. Davis reminded you really how many RAIs we generated.

Well, the point is that geology really is the framework in which everything sits. Of course, I'm a geologist, what else would I say?

But the point is that we really did chew on it pretty hard. We went through the entire section, site, regional, and geology, vibratory ground motions, surface faulting, stability of subsurface materials, and slope stability -- all of those issues are going to be spoken to.

And the applicant again, as Mr. Davis specified, really did identify and assess rather carefully the site and regional geology characteristics and features as, of course, is required.

I want to show this slide just to sort of set the scene really for what the geology is, and sort of point out some key features that were of concern to use that we really did want to deal with.

I would like to point for one thing the Pen Branch fault. I would also like to point out this little blue line that happens to be the outcrop trace of that blue bluff marl that is in fact the bearing unit.

The Pen Branch fault, since it is a structure, was a concern because it is a structure, and that's a key issue, really. But even though there were no OIs, no open items associated with that, it was something that both the applicant and the geologists were really quite concerned about.

Let me just look at that in a bit more detail in a quick slide to sort of show you -- and again, Mr. Davis showed this is a more cartoon-like cross-section.

But this actually illustrates the location of the Pen Branch, and you will note that in fact it will dip beneath the area of the nuclear island. This blue line is in fact the blue bluff marl, so the sequence that people have spoken about, the Atlantic coastal plain set here, you have older rock that's in excess of 200 million years old on both sides of this structure. So this is sort of the geologic setting, and again this feature was proven rather concisely to be a noncapable fault. That is to say it was less than 1.8 million years old. The quaternary is our cut-off.

So that's sort of an introduction to why the geology is important.

I'm going to pass it now to Sarah who will talk about the seismology features.

MEMBER BANERJEE: Does that mean it's not active or something?

MR. STIREWALT: That means it is not active, exactly right. Thank you for the question.

MEMBER CORRADINI: And so the numbers that you quoted were the limits where you do an estimate, but if it's less than this, it's considered active; if it's longer than such, it's considered not active?

MR. STIREWALT: That is correct. If it is older, prequaternary, then it's nonactive, noncapable, nonseismogenic.

MS. GONZALES: I'm going to talk about section 2.5.2 and some of the open items we had, the significant ones.

One of the main review areas for 2.5.2 is the applicant's probabilistic seismic hazard assessment, or PSHA.

The applicant used the 1986 EPRI PSHA size source model as a starting point in its PSHA. This model is comprised of input from six different teams.

This figure shows an example of one of the teams' seismic source characterization, and you can see that there's various areas on the map.

In the central and eastern United States, seismic source -- earthquakes are modeled as area sources rather than discrete fault sources. So that's what those areas correspond to.

I just want to show you these on the map. This is the Vogtle site here.

MEMBER BROWN: Does the green, blue, and orange represent areas?

MS. GONZALES: Yes, those are all different area sources.

MEMBER BROWN: That's fine.

MS. GONZALES: Yes. This is the Vogtle site, and these are all the various source sites that one of the EPRI teams defined, and this is the Charleston source zone that they defined, and here just outside the 200-mile site radius is the eastern Tennessee seismic zone, and way out here is the New Madrid seismic zone.

MEMBER BROWN: So even Charleston is a zone?

MS. GONZALES: Charleston is a zone.

MEMBER BANERJEE: Is that where there was a big earthquake?

MEMBER BROWN: Yes.

MEMBER APOSTOLAKIS: Okay.

MS. GONZALES: And so since the model was developed in the '80s, there's been a lot of updates, you know, various new data since then. So a lot of our review was focused on determining whether the applicant adequately updated the EPRI model to make sure that -- to account for any new information.

MEMBER APOSTOLAKIS: Now the update means a new zone, or different numerical characteristics of the same zone, or both?

MS. GONZALES: Anything. I mean anything, new data, new source zones possibly, or --

MEMBER BANERJEE: Is this sort of an expert elicitation or --

MS. GONZALES: The EPRI model, six or 16, they used input from the six different teams, so that's kind of like an expert.

MEMBER APOSTOLAKIS: It's more of an expert interpretation rather than elicitation.

MEMBER CORRADINI: Right. We had a day of tutoring of this. Remember?

MEMBER APOSTOLAKIS: Yes.

MEMBER BANERJEE: Moving on.

MEMBER CORRADINI: I remember the faces in the crowd, so we must have done something.

MS. GONZALES: So a major update the applicant formed was of the Charleston seismic source zone. Even though the EPRI teams did the final Charleston seismic source zone, new geologic data had been become available, which required an update.

MEMBER STETKAR: I was going to ask, does this include the 2008 U.S.G.S. update?

MS. GONZALES: That's a different model. The applicant kind of looked at that to help it --

MEMBER STETKAR: Did they incorporate it somehow?

MR. MUNSON: Actually -- this is Cliff Munson, the branch chief.

This was done before that 2008 update, but the actual U.S.G.S. 2008 model captures this update that was done for the Vogtle site. So the U.S.G.S., when they updated their 2002 hazard, they used this as part of their update. So it's kind of meshed together.

MEMBER STETKAR: Okay.

MS. GONZALES: So this figure shows their updated model and you can see there are four different source zones that they used to characterize the source zone, and they each had different weights, and the most weight was given to the source zone A, which is this area here.

MEMBER APOSTOLAKIS: So does this mean from the seismic point of view, things are more severe?

MS. GONZALES: Yes, the update incorporated paleoliquefaction data, which I'm going to talk about in the next slide a little bit. But the return periods, since they had this data, the return periods of large earthquakes were shorter than predicted by seismicity, which was used as the -- the EPRI teams were out on seismicity to determine happen large earthquakes occur.

MEMBER APOSTOLAKIS: I remember seeing 100,000 years going down to --

MS. GONZALES: Right, between 600 and 1,000 years, large earthquakes occur at that interval.

MEMBER RYAN: The return date means how -- when it may come back?

MS. GONZALES: Yes. How frequently the earthquake occurs.

MEMBER BROWN: So instead of a longer time, the time was shorter?

MS. GONZALES: It was shorter, yes, so the hazard was higher from that source zone.

MEMBER APOSTOLAKIS: So if I don't much

about earthquakes, I'm a member of the public, what is it that guarantees to me that if I do it again in 20 years, we won't go down to 60 years?

MS. GONZALES: Well, they used what available data they had, and I mean if they find new data, then there's, you know --

MEMBER CORRADINI: No guarantee.

MS. GONZALES: That could change things but, you know, if the record is longer, they could --

MEMBER POWERS: If it drops down to 60 years, then historically that record works.

MS. GONZALES: Well, they had a 5,000-year record that they looked at, and you know, they determined that those time intervals from now.

MEMBER APOSTOLAKIS: Do we have reasonable assurance that the 600 years will not go down to 100?

MEMBER POWERS: Of course, George. If it's 100, then you've got at least two data points and a historical record.

MEMBER APOSTOLAKIS: No, because it's deterministic.

MS. GONZALES: Well, they have a 2,000-year -- the 2,000-year paleoliquefaction record is pretty complete.

MEMBER APOSTOLAKIS: I mean that's pretty serious reduction from 100,000 to 600.

MEMBER RYAN: I think the point is the record of 2,000 years is not spotty, it's fairly --

MS. GONZALES: The paleoliquefaction record is pretty complete.

MEMBER APOSTOLAKIS: So there is some evidence, scientific evidence, that it's not going to go below the current estimate, significantly, anyway?

MEMBER RYAN: That would be low.

MEMBER APOSTOLAKIS: Be low.

MEMBER POWERS: Just tell him -- just explain to George that it's a Bayesian update of the prior created back in the 1980s based on opinion that the liquefaction data has discovered since then.

MEMBER APOSTOLAKIS: I read the document, and it says from 1986 to now, in this period of 20 years, we had this dramatic change in the return period. I'm wondering what's going to happen in the next 20 years. So I shouldn't worry? I mean can you give me an answer?

MS. GONZALES: Well, the 2,000-year paleoliquefaction record, the applicant determined that to be complete, very complete based on their field work, and the return periods are, you know, 600 years, 550 to 600 years.

MEMBER APOSTOLAKIS: So what you are saying is that it is not likely that we will have additional information in the next 20 years with dramatic changes in it?

MS. GONZALES: Not likely unless, you know -- yes, in this new field they dig up a hole that shows something they hadn't seen, but that small area is pretty well investigated as far as --

MR. STIREWALT: And again, that's what the paleoliquefaction does suggest. I mean that's good, strong, solid geologic evidence for that possibly a 500-year or so interval. So that's really nailing it better than we can nail anything else pretty much on the East Coast. That's really a good, solid, strong data point, and really quite good geologic for talking about that timeframe.

MEMBER CORRADINI: So to put it a different way, I guess, just so from the standpoint of assurance, this is one of the places you have high assurance compared to where you might have more uncertainty in other places in the U.S.?

MEMBER RYAN: Correct me if I'm wrong, folks, but to me the Charleston earthquake of 1886 and all the study that's gone into that, and across the state of South Carolina and over into Georgia, is probably more robust than a lot of other areas of the country that haven't been studied at all.

MEMBER CORRADINI: Definitely.

MEMBER APOSTOLAKIS: But people studied in '86, too. It was studied in 1986, also, and yet in the period of 20 years, we have such an enormous charge.

MS. GONZALES: Well, I guess in 1986 they relied on just the seismicity, the historical seismicity that was recorded, so that, you know, there's only one earthquake, the large Charleston earthquake that they could really rely on.

MEMBER RYAN: Well, I think the point is things didn't change, they just broadened their data base, and they made their estimate robust.

MR. CONSTANTINO: The pinnacle of liquefaction studies began in '85, '86, and they mainly had to do with studies associated with the Savannah River site across the way. That's the part that nails it down. Prior to '86, there wasn't good paleoliquefaction studies that were performed, and I think that's really what Sarah is saying, that nails down --

MS. GONZALES: Yes, they relied on seismicity data instead for the return period.

CHAIR SHACK: So we should be suspicious of any place where they're working with just seismicity data.

MR. CONSTANTINO: Yes, that's right. In all of those areas they're really spending a lot more time now on the paleoliquefaction side.

MEMBER POWERS: Let's move on.

MS. GONZALES: So the Charleston record -- I said on the previous slide the Charleston update was based on liquefaction features from historic and prehistoric earthquakes.

Liquefaction features occur in response to strong ground shaking, and this slide just summarizes what liquefaction is.

You can see -- that's strong ground shaking and the soil will become like a fluid, and this fluid can penetrate the overlying layers, and you can get the formation of sand dikes as well as sand blows at the surface, and this also shows a sand dike. And these could get preserved in the geologic record. So that's what the applicant used to help better define the geometry of the Charleston source zone as well as the recurrence interval. They just relied on that data.

MEMBER SIEBER: So if you are looking for these things, you look for features like that and then you have to use some kind of program technique to find out where the fault is and how it's opened up?

MS. GONZALES: For sand blows, they are visible from the surface, that that's a sand dike. Usually they're exposed in the stream of river banks and things like that.

MEMBER SIEBER: That's not the only fact.

MS. GONZALES: Yes.

MEMBER SIEBER: That's not the only form.

MS. GONZALES: No, that's true.

MEMBER SIEBER: It's characteristic of --

MS. GONZALES: Yes.

On the next slide I'm going to discuss briefly the three significant open items we had related to geology and seismology.

The first one has to do with one of the EPRI-SOG teams, the Dames and Moore team. We were concerned with the way that they had modeled their source zones and its effect on the seismic hazard.

However, the applicant -- to resolve this open item, the applicant determined the Dames and Moore contribution to the hazard was very insignificant at the Vogtle site, and this has mainly to do with the dominance of the Charleston hazard, so that that closed that open item.

And the next open item was to do with the the eastern Tennessee seismic zone. The applicant didn't perform any updates on this seismic zone.

However, we were concerned because more recent studies assigned much larger maximum magnitudes to the source zone than the EPRI teams had assigned.

So what we did, we did our own sensitivity study, and we increased the maximum magnitude of the eastern Tennessee seismic zone, and the results showed that the hazard was still insignificant at the Vogtle site. It was too far away to really contribute to the hazard. So that closed that open item.

And lastly, we had an open item to do with the presence of injected sand dikes in the area. We requested more information to ensure that these sand dikes weren't related to earthquakes.

MEMBER BROWN: What is an injected sand dike versus one that's related to earthquakes?

MS. GONZALES: They're both injected, but one is related -- the earthquake can cause the sands to liquefy, whereas the applicant in this case, they provided us with field evidence to show that these sand dikes were to do with collapse and dissolution of the overlying Utley limestone.

MEMBER BROWN: Okay, not earthquake generated?

MS. GONZALES: That's true. That's correct.

MEMBER CORRADINI: In this case then, due to aquifers or just water flowing and dissolving the minerals?

MS. GONZALES: Yes, the limestone was dissolved.

MR. STIREWALT: In this case, they are located in three specific spots at the site. The Utley sort of underlies it, but we know there's dissolution and in fact it's related specifically -- I mean spatially directly related to those little pockets of dissolution.

So the thought is dissolution occurred in the Utley. You had literally collapse of sediments above, and that was the process for fluidizing the obviously water-saturated sediments and moving them.

MEMBER CORRADINI: Okay. Thank you.

MS. GONZALES: So moving on to -- well, that closed that open item, so moving on to section 2.5.4, the main topics here are engineering properties of soils and rocks, site exploration, geophysical surveys, liquefaction potential, and static stability.

So for the Vogtle ESP, we had 12 open items addressing the adequacy of the field and lab testing of subsurface materials. Measurements of shear wave velocity as well as serial degradation properties.

There is also a chronic condition, too, which is added, and that required the removal of the upper sand layer.

And there were also 12 COL action items.

MEMBER BANERJEE: And the shear wave velocities were measured in situ? Or in samples or?

MS. GONZALES: They were rated in situ.

MEMBER BANERJEE: In situ.

MEMBER ARMIJO: Just a quick question. Was the permit condition a result of the RAIs, or did the applicant come forward initially with the intent of removing the upper sand layer?

MS. GONZALES: Well, the upper sand layer was susceptible to liquefaction, so the applicant's liquefaction analysis assumed that that upper layer was removed, and the site response analysis also didn't include that upper layer as well. So --

MEMBER ARMIJO: Okay. So they intended, when they made their application, to remove it.

MS. GONZALES: Yes.

MEMBER ARMIJO: You just made --

MR. ARAGUAS: That was their proposal.

MEMBER ARMIJO: Okay.

MS. GONZALES: And so as part of the LWA, the applicant provided us with a significant amount of additional data, which is shown in this table here, and this additional information was sufficient to resolve the open items and the COL action items.

CHAIR SHACK: I'd say that's more data.

MS. GONZALES: That's it.

MR. MUSICO: Good morning. My name is Bruce Musico. I'm a senior emergency preparedness specialist within the Office of Nuclear Security and Incident Response, NSIR, and I am here to just briefly go over the Vogtle early site permit application and the complete and integrated emergency plan that they propose for the site.

As you can see in the first slide, the emergency plan that Southern proposed for the Vogtle site is unique in that it is the first time that a complete emergency plan has been proposed under the new Part 52 licensing process.

What they proposed again was a complete and integrated emergency plan which in essence means it includes the on-site plan as well as the off-site plans, which consist of the state plans and local county plans.

We conducted this review in conjunction with or along with FEMA. FEMA actually reviewed the offsite emergency plans, and they did a very detailed review.

We looked primarily at the on-site, but we also reviewed the offsite as well, just to see how it was integrated with the on-site plan.

So when you see the term "complete and integrated emergy plan," that means the on-site and offsite plan working together.

Also this was a first-of-a-kind use of the ITAAC, emergency planning ITAAC -- inspections, tests, analysis, and acceptance criteria. And a number of the RAIs that we were asked were associated with this first-of-a-kind use of ITAAC, and we were able to resolve those.

In the SER with open items, the initial SER with open items, we identified 13 emergency plan open items and three COL action items.

In the subsequent advanced SER, which is what we are looking at now, all of the open items were closed, resolved, and the three COL action items actually were eliminated and change into permit conditions.

On the next slide of interest is open item 13.3-4, which dealt with emergency action levels, or EALs. This is unique here because in our review of the emergency plan, it is utilizing the EAL scheme that will be endorsed in the Nuclear Energy Institute's document, NEI 0701, which is a document that will reflect advanced passive reactors consisting of the AP1000 and the ESBWR.

We are currently reviewing NEI 0701 for endorsement, but that's a work in progress right now. That is a parallel dependent licensing action that this application is dependent upon. Hence we have permit conditions associated with it.

In addition, we have an ongoing parallel licensing dependent action in the AP1000 design control document, DCD, in which Westinghouse has submitted proposed amendments to the AP1000 certified design, some of which are EP related, primarily the location of the technical support center, and that is undergoing current review by the NRC in the context of a rulemaking proceeding.

So, again, we have two parallel dependent licensing actions that we had to accommodate and consider in the review of this application because it's dependent upon those two. Hence we have permit conditions; we also have ITAAC.

As I said before, the EAL resolution -- the EALs had been resolved by the introduction of six permit conditions, 2 through 7. Specifically we have three sets. We have 2 and 3, which deal with NEI 0701, 4 and 5, the AP1000 amendments, 6 and 7 deal with a broader description of the requirements for EAL, but basically it parrots what's in Appendix E of 10 CFR Part 50.

The reason there's two for each one is that we have separated them out where unit 3 applies to permit condition 2, 4, and 6, and unit 4 is permit condition 3, 5, and 7.

And that fully covers the requirements for emergency action levels, but to add an additional assurance, we also proposed an ITAAC 1.1.2, which requires a complete EAL scheme be eventually developed that's consistent with Reg Guide 1.101.

If you're familiar with Reg Guide 1.101, that is the vehicle by which we endorse various documents such as NEI 0701.

And so when NEI 0701 is endorsed by us through that separate ongoing proceeding, we will issue a revision to Reg Guide 1.101 which will endorse it.

So six of the seven EP permit conditions relate to the emergency action levels. We have a final permit condition that relates to the location of the technical support center.

This was interesting because the AP1000 DCD amendments above, one of the amendments has to do with changing the characterization of the TSC location in the certified design from a tier 1 information item to a tier 2*. And I won't get into the details. Those are defined in Appendix D of Part 52. But in essence it eliminates the need for a subsequent COL applicant to submit an exemption request with the COL application. They merely request a different location for the TSC in the COL application, which is what we have here for the Vogtle application.

So that has to do with the TSC.

With respect to the early site permit, we made it a permit condition that the COL applicant will subsequently have to resolve the difference between the application, which says we're going to have a TSC separate from that which is identified in the AP1000 design, yet identifies the AP1000 certified design, which identifies the TSC being in the annex building close to the control room. So we have a conflict there.

So we identified that as a permit condition and we are actually reviewing the COL application that's come out in this regard, and the applicant has proposed a departure from the AP1000 certified design to accommodate this.

But the ESP itself basically approves the relocation of the TSC subject to resolution of this conflict in the subsequent COL application. So that's what the TSC is about. We don't have a problem with that.

Now a question was brought up yesterday with respect to the two-minute walking distance. I'm not going to get into a lot of detail here, but I made reference to NUREG 0696, which is the applicable guidance document, 1981 guidance document, which recommends an approximately two-minute walking distance between the technical support center and the main control room.

We had quite a lengthy discussion yesterday with respect to the precedence of allowing a TSC, approving a TSC here, that's farther away. And I made reference yesterday to a prior licensing action that I couldn't recall the plant, but it had to do with our approval of a TSC located 15 minutes away from the main control room.

I subsequently dug up the case. It was for the Clinton station. It was a January 19, 2007 safety evaluation report that we wrote. It's a nonpublic document, ADAMS No. ML070110425, entitled "Subject: Relocation of the Technical Support Center for the Clinton Power Station Under Docket 50-461."

The purpose was the proposed change would relocate the technical support center from its current location adjacent to the main control room to the training facility on the east side of the owner-controlled area.

In our analysis -- and this was done by a different reviewer, and I was not aware of this study at the time, this safety evaluation report at the time when I reviewed and approved the relocation for Vogtle. But it turns out we were consistent.

In fact, the reviewer had brought it to my attention and said, well, you didn't know about this, but we are still on the same wave length here. And I sort of was glad to hear that. I was glad to hear that.

MEMBER SIEBER: That's not the only plant that's had to take exception.

MR. MUSICO: And this was given as an

example, not the only example, but one example of prior approvals that we have given with respect to relocation of the technical support center.

Specifically it says the transit time between the proposed TSC and the main control room will be approximately 15 minutes and includes time to traverse through the security barriers and it also mentions the two-minute walking distance.

However, it comes to the conclusion, "While the transit time is greater than the recommended NUREG-0696, the enhancement to the communications and instrumentation as well as the enhancements based on an increase in the physical size of the TSC is an acceptable alternative to the functional requirements of NUREG-0696 and is acceptable."

So the precedent has already been set in prior evaluations that we have done.

So the Vogtle application was merely consistent with what we have approved in the past, so it wasn't necessarily precedent setting. So I just wanted to clarify that and bring that to your attention.

Okay. And then finally, as I said, with respect to the EALs, we identified an ITAAC. The four bullets correspond to the four columns that are in the ITAAC table.

The first bullet merely is the first column, which parrots 10 CFR 50 Part 47(b)(4), which deals with the requirement for a standard emergency classification scheme, which includes the four classifications as well as the EAL, associated EAL scheme.

The second bullet merely parrots the evaluation criteria in NUREG-0654(d)(1), which again talks to the EAL scheme required. And you will see there that in the first bullet, it says "the basis of which includes facility system and effluent parameters."

Now these are specific as-built aspects of the reactor, details that you don't have to solve yet until you build the building, the reactor and the systems. And so that's why we need placeholders to accommodate that, hence the permit conditions in ITAAC. So ITAAC was well suited for this.

Secondly, the second bullet, the specific instruments, parameters, and equipment status shall be shown.

And finally, the plan shall identify the parameter values and equipment status.

Again, these are dependent upon the as-built aspects of the plant.

And then the final two bullets are the most important here, in that the ITA inspection, tests, and analyses says that the analysis of the EAL technical basis will be performed to verify as-built site-specific implementation of the EAL scheme, and acceptance criteria is the EAL scheme is consistent with Regulatory Guide 1.101, which is expected to be endorsed -- which is expected to endorse NUREG-0701 following the staff's review.

And that's it in a nutshell. Any questions? Thank you.

MR. ARAGUAS: All right, that brings us to our discussion on LWA.

All right, so just to reiterate Jim's comments on the activities that have been requested as part of the limited work authorization, they have asked for placement of engineered backfill on the site. They have asked to place the mechanically stabilized earth retaining walls. They have asked for placement of links to be backfilled, mud mats, and water proofing material, and with that we will speak to each of those, and with respect to the area that we reviewed.

MR. WANG: Okay. My name is Weijun Wang, geotechnical engineer of the NRO. I will briefly present the staff review on LWA request, section 2.5.4, the subsurface material in the foundation.

The staff issued 26 RAIs addressing the concerns about the LWA request. Basically the first one is the adequacy of the borings at the site. Because during the ESP, the applicant performed 14 borings and only three of them to the load-bearing layer, which is blue bluff marl. And because of that, and also that 14 borings did not cover the whole footprint of the AP1000, so that's one concern of staff.

And the second concern is the adequacy of the determination of engineering properties of the subsurface materials. To continue, they had 12 tubes of samples for the laboratory test, again for the load-bearing layer. So that is not sufficient to provide us reliable soil properties.

We are sort of concerned with the adequacy of the DAC field soils. Through the ESP, the applicant did not provide the most details about the backfill, especially about the property of the backfills. It's all the parameters were assumed or used from based on unit 1 and 2 site investigations.

We found staff concerns. The applicant did the following. Regarding the further concern, the applicant come back additional 174 borings, with 42 borings -- the 44 borings penetrated into the blue bluff marl layer. Again, that's load-bearing layer. And also 70 borings cover the footprint of the unit 3 and the 4, so which sufficiently provided enough borings for the site investigation.

So this responded to our concern. You can see because they conducted a lot more borings. They collect a lot more samples, too. And also they performed more field tests. For example, for the load-bearing layer, the blue bluff marl, they performed 742 SPT measurements. And also 94 soil samples.

And for the lower sand layer, they measured 111 SPTs, and another 29 undisturbed samples. So because of that --

CHAIR SHACK: That's that 1,000-foot coastal plain sediment, is that what you mean by the lower sand stratum?

MR. WANG: The lower sand stratum is underneath the blue bluff marl. So because the applicant collected more samples and come back with more laboratory tests, and they provide a lot more data to determine the engineering property, including the static and the dynamic property of the subsurface material in the site. They responded to our concern about backfill. Besides providing the detailed information about backfill material, the applicant also provided a two-phase test path program. The test path program, we provide the in-field property of the backfill.

In addition to that, they also provide what developed ITAAC for the backfill. The ITAAC, you can see concentrates on two major parameters. One is the minimum compaction, the minimum 95 percent compaction of the backfill. Another one is the minimum shear wave velocity, which is 1,000 feet per second.

These two major parameters will ensure that the backfill soil property will meet the design requirements. In turn, it will ensure the stability of the material underneath the foundation.

Based on staff review of the applicant responded to our concerns, and also based on the two site audits conducted by staff, one was in December '07, another one in July of this year, the staff concluded the applicant adequately answered our concerns. Therefore, the 26 RAIs were resolved.

Any questions?

MR. TEGELER: Good afternoon. My name is Bret Tegeler, and I work in the structural engineering branch in the Office of NRO. I am here to present with John Ma and Carl Constantino, our consultant, the structural engineering review of the LWA application.

To start off, as you recall, the applicant is requesting to place foundation supporting elements that will eventually -- that will be below the nuclear island base mat.

As a result of that, the structural engineering branch reviews the seismic demands on those elements, namely, the reinforced -- I'm sorry, the concrete mud mat and the waterproof membrane.

So the three SRP sections that we performed our review under are 3.7.1., 3.7.2, and 3.8.5.

The primary, if you will, finding resides in 3.8.5, the foundation, the assessment of the sliding and overturn stability of the nuclear island base mat placed on these elements.

To support that 3.8.5 finding, we need to determine the seismic loads, which are fed from 3.7.1 and 3.7.2.

I'll just quickly cover it. In 3.7.1, our primary goal or review is of the site GMRS, the site SSE, and we take that and compare that to the design response spector for the AP1000.

And that's the next slide here.

In 3.7.2, we essentially review the seismic analysis or soil-structure interaction essentially looking to make sure and verifying that the results make sense for use in the 3.8.5 evaluation.

I alluded to this one. This is a comparison of the Vogtle site GMRS, essentially the ground motion response spectra, which becomes the SSE for the site.

The GMRS exceeds the AP1000 design response spectra in essentially two frequency ranges, below 1 hertz, and a higher range above 7 hertz.

As a result of the exceedance, the applicant has to perform site-specific soil-structure interaction to demonstrate that while though you have an exceedance, a site-specific analysis is required to demonstrate the -- sort of the appropriateness of the site for the AP1000 design.

MEMBER SIEBER: That analysis was done and you have it?

MR. TEGELER: Yes. That was performed by

the applicant and it was submitted.

MEMBER SIEBER: And your conclusion is based not so much on that as on the analysis?

MR. TEGELER: Our conclusion -- this -- these data, if you will, are input into that SSI. The SSI results reflect these data.

MEMBER APOSTOLAKIS: I can't hear you very well.

MR. TEGELER: I'm sorry. The applicant's 2-D SSI analyses that were performed reflect these design -- these spectra.

MEMBER APOSTOLAKIS: So even the spectrum is different --

MR. TEGELER: That's right.

MEMBER APOSTOLAKIS: -- it's about -- what matters ultimately is what?

MR. TEGELER: What ultimately happened, that your loads and in-structure response spectra are below the design basis. I'm sorry, I should say with design basis.

MEMBER SIEBER: And if I look at that, that doesn't tell me that.

MR. TEGELER: That's because this is just a first input. The actual -- the first piece of the analysis is a comparison of your site to the certified design response spectra.

If you exceed that, that's not the end of the day. That doesn't mean that you can't stick an AP1000 on the site. What it means is you have to do site-specific analysis to make that demonstration.

MEMBER CORRADINI: Which you said earlier, and I didn't hear, has been submitted to you?

MR. TEGELER: Yes.

MEMBER CORRADINI: Okay.

MR. TEGELER: This does not reflect -- this slide itself is not a result, if you will, of the 2-D analysis. This is a site -- this is the response of the site which drove the applicant, because they had a site exceedance, drives them to do a site-specific SSI analysis.

MEMBER SIEBER: Do you have similar graphics that would show the result of this site-specific analysis?

MR. TEGELER: I have a back-up slide that may help that. Maybe right when I --

MEMBER SIEBER: Yes, when you get to the end of your portion, I'd like to see that, because right now the only thing I'm convinced of is why we ought not put that --

MEMBER APOSTOLAKIS: Is this site

specific?

MR. TEGELER: This is site specific in that the site GMRS, that blue line, is a global site condition.

CHAIR SHACK: The red line is the certified spectra, right?

MR. TEGELER: That's correct.

CHAIR SHACK: That's not site specific, that's --

MEMBER SIEBER: That's the design.

MR. TEGELER: So recall for Rev. 15, that's the hard rock design, the Rev. 16, now 17, is going to be a soil design, and so you -- in addition to this -- these -- in addition to comparing to just the Reg Guide 160 spectra or the red line, you also look at the soil profile, so that's the other --

MEMBER ARMIJO: Does it follow that the loads as a result of -- would be different or you'd have greater loads on the structure, the structures and the components in the plant? Would the actual spectra then be bounding curves?

MR. TEGELER: What happens is you are, for this case, you have exceeded the design at the -- in the far field. When you do your SSI analysis, you know how the nuclear island is embedded in a soil column that's 1,000 feet deep. You have a lot of energy dissipation due to soil damping, radiation effects from the structure itself compared, and so what I'm saying is I'm jumping ahead to the results of the SSI analysis, but they demonstrate that while you have the exceedance at the site, because you have your embedded 40 feet in this 1,000-foot soil column, you have a tremendous attenuation and you don't see the exceedances to this degree in the in-structure.

MR. CONSTANTINO: Can I say something, Bret?

MR. TEGELER: Thank you.

MR. CONSTANTINO: The design is based on that design, the red spectra, put onto a number of different sites, and it's the envelope of all of those. And now we come to particular site-specific evaluation. If the spectra fall below the design it is clearly no problem. If it exceeds, there's still probably no problem because of the detailed site-specific characteristics which are different than the envelope.

So that's really -- the 2-D results show that, and we still have this open issue which we are doing 3-D to confirm that.

But the basic idea is even if we have

exceedances, we don't necessarily have exceedances in structure demands on equipment and the structures.

MR. TEGELER: Thank you.

MEMBER SIEBER: But is the item closed, in your mind?

MR. TEGELER: It's not closed in that we're -- that's an ongoing review.

MR. ARAGUAS: But let me jump in and clarify. It is closed for the LWA, and we've got to be clear about that. It is closed for the LWA. We have found that 2-D is acceptable, but for the purpose of in-structure response, 3-D is what we're asking for, and that's being reviewed as part of the COL. But that's separate from the LWA.

MEMBER SIEBER: On the other hand, a modification, which basically was designed for, which the LWA is issued, could be a potential fix for this issue. You know, you go deeper, pull in better soil, but it is deemed not to be necessary at this time?

MR. TEGELER: That's true.

MR. CONSTANTINO: There is no such as bad soil. Unless we go to concrete or something, and we don't want to face that issue.

MR. TEGELER: So essentially, I'll wrap up our findings in 3.7.1.

We found that the proximate method for developing the FIRS at the site was adequate, and that the FIRS also satisfied the regulatory requirements for a minimum spectral change for the foundation.

3.7.2, again, we just talked about seismic fouling. We did find the use of the applicant's 2-D models appropriate for developing the seismic demands for the purposes of sliding and overturning stability.

And just as a check on that, we compared those, the applicant's results, to the soft soil case of the AP1000 design.

And as I mentioned, these demands are then passed to 3.8.5, for the actual stability evaluation conducted under John Ma.

MR. MA: My name is John Ma. I work for the structural engineering branch of NRO.

As you have been presented by the applicant, the last slide showed you the boundary of the MSE, the wall. That boundary wall will be the boundary for the base mat of the nuclear island structure.

So at this stage we are only saying you can do the base mat preparation work, not including the base mat. So the base mat is what we call structural foundation for the nuclear island structure.

So initially they came in, they want to do the foundation. They want to put the rebar cages assembly for foundation. We said no, no, no, you cannot do that because we have not resolved many RAIs with the Westinghouse AP1000 base mat yet. So they withdrew that, so as of now the only two preparation work for the base mat is the mat, mud mat, and membrane. They want to put a membrane between the mud mat.

The thickness of the mud mat is 12 inches, so they want to put the waterproofing membrane between the mud mat, which we said if you want to do that, you'd better show me the result of the mud mat will not slide, the upper portion of the mud mat will not slide relative to the lower portion of the mud mat during SSE.

So they went out, they got the test result and showed us, I think just the first -- yes, the first one they said it's the coefficient of friction of .7 or greater, and they showed us those test data.

And the second we got from the applicant is their test result of their soil with a coefficient of friction .45.

Now because this .45 is lower than .7,

therefore the membrane will not control the sliding effect. So the soil will control.

MEMBER CORRADINI: This was, I'm sure, discussed yesterday, but just so I understand what you just said, so they're putting in the mud mat, the membrane is going inside the mud mat.

MR. MA: Yes.

MEMBER CORRADINI: That's the last thing they are allowed to do ahead of time per your approval.

MR. MA: Yes.

MEMBER CORRADINI: And the soil that's naturally there falls below that, unnaturally there -- no --

(Laughter.)

MEMBER CORRADINI: The backfill soil is what you're talking about, and it's that interface between that backfill soil and the mud mat that you're talking about.

MR. MA: Yes.

MEMBER CORRADINI: Thank you.

MR. MA: And the third data we got from the applicant is the dynamics bearing capacity of the soil is 42 kips per square foot.

Now let's go to the second slide now.

As you recall, Bret told you, their soil motion input exceeded the AP1000 design spectra. Because of that we said -- we asked the licensee -- the applicant, because you exceeded already, now you got to show me during an SSE your base mat or your nuclear island structure will not slide during an SSE.

So they made a calculation, the inertial force during the SSE are large. They also made a calculation the frictional forces, which use the .45 coefficient of friction multiplied the total dead weight of the nuclear island structure.

As you can see here, the frictional force is greater than the inertial force. Now, remember, this inertial force is calculated based on the site-specific spectra, so from this site has nothing to do with AP1000 itself.

CHAIR SHACK: Except for the weight.

MR. MA: Except the weight, yes. Sure. Okay.

And now let's go to the next slide.

A more important phenomenon in structure during an earthquake is the break into the soil, and some of you may have seen, especially in Japan, during an earthquake, the whole building just turns over, some totally, but some just, you know, go down and will not come back anymore.

The reason for that is because the dynamic pressure on the soil exceeded the soil capacity. So it goes down.

So we asked the applicant, you calculate that number for us. So they did. As you can see, the first line, they said the maximum dynamic soil pressure during an SSE, the greatest one is under the nuclear island, which is 17.95 kip per square feet.

In the first slide they already told us the soil capacity is 42 per kip per square feet. So if you divide 42 kip per square feet by 17.95, you get a safety factor of more than two.

So in this case we know the footprint, right now they have it under their MSE wall. In the future when they build this plant, according to AP1000, according to their weight, just like I mentioned, if their weight is correct, we know it will not slide, we know it will not overturn in an SSE. And therefore we said for the LWA, this is okay.

MEMBER BROWN: Did anybody validate their calculations?

MR. MA: We did the same calculation, and this -- we wanted to make sure, you know, their calculation is in the right part, so we made our own hand calculation.

MEMBER APOSTOLAKIS: Was it on the back of an envelope?

MR. MA: What? Well, no, not really, and very close to their number.

MEMBER BROWN: Okay. So you came out approximately what they came out with?

MR. MA: Yes. Yes, approximately the same. So we think that's good enough. And with this kind of, you know, safety factor, over two, we think that's good enough.

Thank you.

MR. ARAGUAS: Okay, so that brings us to our last slide, the conclusion slide, and if you'll bear with me, I'll just quickly read through the conclusions for both the advanced SER and the LWA.

MEMBER STETKAR: One thing, Christian, let me just -- the slide before that, and I know we're short on time, but you said nothing about the in-plant equipment response. That's still an open --

MR. ARAGUAS: Correct. That's part of the COL.

MEMBER STETKAR: The COL?

MR. ARAGUAS: That's correct.

MEMBER STETKAR: That's fine.

MR. ARAGUAS: Okay. So for the SER and the LWA, the conclusions, you can tell, are shared by the two.

I would say the only area where it's different with the SER with respect to the LWA is with the LWA we're not approving site characteristics or design parameters in terms of conditions as part of the LWA. So that's totally specific to the SER.

But having said that, I'll read through these.

The application meets the applicable standards and requirements of the act and the commission's regulations.

Site characteristics, site parameters, and terms and conditions to be proposed to be included into the ESP meet the applicable requirements of 10 CFR Part 52.

There is reasonable assurance that the site is in conformity with the provisions of the act and the commission's regulations.

The proposed ITAAC, those being the emergency planning ITAAC and those associated with the LWA, are necessary and sufficient and within the scope of the ESP, provide reasonable assurance that the facility has been constructed and will be operated in conformity with the emergency plans and LWA, the provisions of the act, and the commission's regulations.

And lastly, the issuance of the permit and the LWA will not be inimical to the common defense and security or the health and safety of the public.

That concludes our presentation.

MEMBER POWERS: Thank you very much.

Do members have any additional questions?

Okay, we will draft a letter conforming with the DAC provision submitted by the subcommittee.

CHAIR SHACK: We will recess for lunch until 1:30.

(Whereupon, at 12:24 p.m., the committee was recessed, to reconvene at 1:30 p.m.)

A-F-T-E-R-N-O-O-N S-E-S-S-I-O-N

1:30 p.m.

CHAIR SHACK: I'd like to come back into session. Our next presentation will be on a review of staff activities associated with potential revisions for 10 CFR 50.46(b) and Sam will be leading our discussion.

MEMBER ARMIJO: Thank you, Mr. Chairman.

On the second of this morning, the Materials, Metallurgy and Reactor Fuels Subcommittee met with the staff and with members of the industry, representatives of the industry. It was a full-day meeting. We went into a lot of detail on this subject and there was some new experimental information. There was also a concept, an approach, that the staff wants to present to us and did present to us on a rule-making process which I believe is both practical and efficient if it's followed.

There is general agreement or growing consensus on the major phenomena that are involved in cladding embrittlement, the dominant role of hydrogen in controlling the embrittlement phenomena, obviously the benefits of having zirconium alloys with very low hydrogen pick-up rates during normal oxidation, normal operation. The validity of testing hydrogen-charged unirradiated specimens to obtain properties equivalent to those of irradiated fuel cladding I think is very valuable information. There's recognition that low temperature quenching after a LOCA or slow cooling after a LOCA has some beneficial effect. The fact that some highest-burn-up fuel can't achieve the very high temperatures such 1200 degree Centigrade compensates somewhat for the hydrogen pick-up during normal corrosion that you find in high-burn-up fuel. Anyway, a number of things. There's convergence.

The industry still has issues on some of the phenomena that were being discussed and the importance that the staff ultimately would place on those phenomena, but I think even there was some flexibility and a very good rule could be written and I think with that I'd like to turn it over to Paul Clifford and he can give us a summary review. We have 25 minutes allocated for Paul and 20 minutes for the industry and we're going to try and hold to our time.

Paul.

MR. CLIFFORD: Thank you, Dr. Armijo.

As he mentioned, my name is Paul Clifford. I am in NRR. I'd like to begin by stating the Office of Research has recently completed a momentous research program which has truly expanded our knowledge of fuel performance during loss of coolant accidents. As you are aware, a description of this research program along with key findings as documented with a NUREG/CR-6967 and RIL-0801.

I'm here today to provide conceptual changes to the rule. The structure of the rule is being revised to provide an optional flexibility in response to some comments we received from the industry and the structure that's really still being developed as we speak. We're in the early stages of this process. It's a long path forward. As such, things like specific rule language is still being developed.

At this time, the staff does not need a written response on our draft strategy or conceptual rule changes. However we welcome any comments from this body. As the rule-making matures, the ACRS will have an opportunity to weigh on the specifics of a future proposed rule and, with that, I will begin.

Following Commission directive, the staff was tasked with developing a performance-based rule which enables licensees to use advanced cladding alloys such as M5 without the need for an exemption. Specifically, we need to expand the applicability beyond the words "beyond Zircaloy or ZIRLO" and that text actually appears at 50.46.

In parallel, a second objective was to capture the results of a high-burn-up LOCA research program and, as I described, in NUREG/CR-6967 and RIL-0801. This research identified new embrittlement mechanisms which were beyond the known phenomena when the rule was written back in 1973.

MEMBER RAY: And that's independent of the desire for performance-based rule-making, the last thing you said.

MR. CLIFFORD: It really is. It really is two sided.

MEMBER RAY: Well, that wasn't clear a couple of days ago as you've made it now. I just wanted to be real clear.

MR. CLIFFORD: Okay.

I'm going to run through each of the changes in the structure of the rule. First is the Applicability. The current regulation in paragraph (a)(1)(I) limits the applicability of "zircaloy" or "ZIRLO." The research included a wide variety of zirconium alloys, ZIRC 2, ZIRC 4, ZIRLO, etc., the results of which have been shown in many cases to be alloy-independent in some phenomena. In other phenomena, we're developing specific test procedures to evaluate the effects of things such as burn-up and alloy composition.

Expanding the rule has no impact on plant safety. I'll set that straight and the strategy is to replace the terminology "zircaloy" and "ZIRLO" with something more generic such as "an approved zirconium alloy." And consistent with current practice, the applicability of any generic criteria within the rule will have to be demonstrated by testing for any new alloy further down the road.

Peak Cladding Temperature, the criterion is provided in paragraph (b)(1) of 50.46 and today it's limited to 2200 degrees Fahrenheit. The research findings show that post quench ductility dramatically decreases if the sample is oxidized in steam above 2200 degrees Fahrenheit. As such, it confirms the current limit of 2200 degrees, that ceiling, and right now we don't intend to change or increase that limit above 2200 degrees Fahrenheit. There's no plant safety because we're maintaining the same criterion and there's no change.

MEMBER APOSTOLAKIS: When you say "beyond 2200" I mean how far?

MR. CLIFFORD: No, we're not changing it beyond 2200.

MEMBER APOSTOLAKIS: In the --

MR. CLIFFORD: The testing that was done at 2200 shows that only after brief period does the cladding become embrittled. So the 2300, timing would be so short. So it really wouldn't be practical.

Local Oxidation, I have a couple slides on this one. Paragraph (b)(2) specifies a limit on local oxidation of 17 percent ECR. New embrittlement phenomena were identified during this research.

MEMBER ARMIJO: Paul, you might want to tell the rest of the staff what ECR means.

MEMBER APOSTOLAKIS: Thank you, Sam.

MEMBER ARMIJO: Since I heard you this morning.

MR. CLIFFORD: ECR stands for Equivalent Cladding Reacted. It's the amount of zirconium metal that's converted to oxide. The research that was done at Argonne identified new phenomena associated. It showed a synergistic effect between oxygen diffusion within the base metal or into the base metal and pre-existing hydrogen. As such, the 17 percent criterion that's currently specified is not always adequate to ensure post quench ductility.

In addition, they determined that the -- Let me go back. Ten years ago, we issued information known as 98-29 stating that licensees should subtract the initial corrosion oxide layer in new ECR from the 17 percent. So if they had a equivalent cladding reacted, if they had 20 mils which is equivalent to say four ECRs, then their limit wouldn't be 17. It would now be 17 minus 4 which would 13. But the research findings show that even that correction was not always adequate to ensure post quench ductility.

MEMBER SIEBER: So we can conclude from that that the current rule was not conservative in all cases.

MR. CLIFFORD: That is correct.

MEMBER SIEBER: With regard to oxidation.

MR. CLIFFORD: That is correct.

MEMBER SIEBER: And a change is necessary, right?

MR. CLIFFORD: That is correct.

MEMBER SIEBER: Thank you.

MEMBER RAY: Both Jack and I have gotten here to underscore that point.

MR. CLIFFORD: Now here's a slide that the subcommittee didn't see on Tuesday and I think it was stated that we wanted to see some more information on plant safety. What I've done is I converted the allowable ECR or measure the ECR to the brittle transition which is a function of hydrogen. I converted that to local burn-up using oxidation properties of each of these with these different alloys and, as you can see, the measured brittle transition for these particular alloys was above the current 17 percent.

However, as you introduce initial hydrogen during normal operation, this isn't hydrogen during the transition. It's initial hydrogen as result of normal steady state corrosion. The allowable ECR would decrease such that as you move up in burn-up it would drop below the current 17 percent.

Now with respect to plant safety, during normal operation, time and temperature you built an oxidation layer and there's some hydrogen that's up-taken up into the metal which the research shows has a direct impact on allowable ECR. But during the buildup of corrosion you're also depleting U-235. So as you burn up the rod, the rod power decreases and we're just showing this. As for fresh fuel, you're allowed a higher ECR but you're at a higher burn-up or I should say a higher rod power and as your allowable ECR was to diminish with burn-up or actually with hydrogen pickup, the rod power would come down.

This is the TMOL. This is the thermal

mechanical operating limit for a typical BWR. It just shows you the tradeoff between rod power and allowable ECR.

Now the strategy for revising the local oxidation, the 17th percent, is there's going to be two options or two alternatives I should say. The first alternative would be that a generic post quench ductility criteria would be specified within the rule. This would replace the constant 17 percent that's in there now and this would be based on the Argonne results. Here's just an illustration. I'm not sure what the lines would look like, but it would be something like this. And, in addition, the licensees or the fuel vendors will have the option of using this alternative approach to showing compliance which would be to run a specific test program for defining specific criterion for their alloy and I have a few slides later on that show what flexibility this allows the industry and I'll get to that.

Now the next issue identified at Argonne was ID Oxygen Diffusion. The research identified that if there's a fuel bonding layer present on the cladding ID that the oxygen can diffuse into the base metal from the ID and hence embrittling the cladding much faster than if there wasn't oxygen diffusing from the inside, if there was only oxygen diffusing from the OD. There are no current regulations on this.

With respect to plant safety, current methods already require double-sided oxidation within the balloon region. So there wouldn't be a significant difference in the limiting time and temperature between the balloon region and outside the balloon region. And also since this a burn-up phenomenon you're not going to get a fuel bonding layer until you're at mid to high burn-up and by the time you were to obtain a fuel bonding layer rod powers, you could have depleted the U-235. Rod powers would be lower and hence there would be much more benign transient for that rod. The strategy is to introduce a new requirement within the rule which requires the licensee to specifically account for ID oxygen diffusion if a bonding layer exists.

Another new phenomena is Breakaway Oxidation. Now the research identified this is a new embrittlement. Essentially it involves the protective tetragonal oxide transforms into an unstable monoclinic structure and as the monoclinic structure degrades it allows hydrogen to be taken up by the base metal which promotes embrittlement. But all zirconium alloys will undergo this transformation in the oxide layer. The question is timing. When does it occur and the requirement -- Timing is very important and one of the key findings was that the timing was sensitive to the manufacturing process. There are no current requirements on this phenomena.

With respect to plant safety, currently for domestic alloys the time for which transformation occurs exceeds 3,000 seconds and we feel that the current LOCA analysis even though they're conservative coupled with reasonable operator actions show the duration that any fuel rods are at these elevated temperatures is beyond the timing of breakaway oxidation.

The strategy would be to introduce a new performance requirement.

MEMBER BROWN: You said it beyond the timing. You meant below?

MR. CLIFFORD: Below, sorry.

The strategy for breakaway oxidation is to introduce a new requirement within the rule that would require specific testing on each vendor's alloy to establish what the measured breakaway time was and then the rule would require that the analysis show that the rods aren't at elevated temperatures up to that point up to that established measured time. And we are also considering required periodic testing which we're sure that changes in the fuel fabrication shop don't introduce changes in the behavior of the fuel which it causes breakaway time to decrease.

We have -- There's a lot of challenge with writing a performance-based rule which meets both the rule-making objectives and also satisfies legal requirements for a specific enforceable criteria. In addition --

MEMBER APOSTOLAKIS: Can you tell us why?

MR. CLIFFORD: Well, the rule is going to be somewhat complex because there's going to be -- Right now, it's just says 17 percent. No exceptions. Now you're going to put in the hydrogen-based criterion for local oxidation and an alternate approach to meeting the rule which would be a required test part which would be an optional test program. So the rule becomes much more complex because there are options essentially in the rule. That makes it a little more difficult to wrote and to satisfy. And another reason, too, is --

MEMBER APOSTOLAKIS: Longer, but I don't know about --

MEMBER ARMIJO: Yes, it could be longer. It would not be so simple.

CHAIR SHACK: The other thing, George, to reflect is we actually wrote a letter one time that told them to put in a performance-based rule. We told them the performance was just to maintain ductility and that one they really objected to on the basis of specificity because there was really almost nothing that was sort of enforceable about it. I mean that was our recommendation when we last wrote a letter on this topic was to have this performance-based rule and that was one of the things that they came back with.

MEMBER APOSTOLAKIS: It could take longer, but I mean difficult to split.

MR. CLIFFORD: Well, it is. Another reason is because for instance you're defining a test that needs to be done. Now you have to define certain aspects of the test which would be legal requirements, but you need to give flexibility for the laboratory to figure out actually how to run the test.

MEMBER APOSTOLAKIS: They could just tell them what the test results should be.

(Laughter.)

MEMBER RAY: George, maybe I can -- One of the debates is over whether periodic testing or not is needed. Paul mentioned, for example, the effect of manufacturing process. You don't really know what the effect is. So that's why they're talking about periodic testing. Well, there's an objection to periodic testing because it seems so superfluous or unnecessary and unjustified. So things like that they get debated.

Do you, for example, prescribe -- How do you define a change in the manufacturing process that requires a new qualification test to be done? Those are the issues he's talking about.

MR. CLIFFORD: And this slide shows, for instance, in the regulation, in the rule itself, you need to specify what post-quench ductility means. What is your figure of merit for post-quench ductility?

MEMBER CORRADINI: So can I just get back to one thing that Bill mentioned just to -- But I thought the objection at the time for that was that they didn't want a qualitative rule even though details would be in reg guide.

CHAIR SHACK: It's the same question about enforceability and in this case now they've moved it off to the test and what's an enforceable testing requirement in the rule level without getting into the reg guide level with detail of --

MEMBER RAY: Without being over

prescriptive.

CHAIR SHACK: -- how you run a test.

MR. CLIFFORD: And like in this slide right here you try to define what ductility is. You're saying you would have to expand this bullet quite a bit.

MEMBER ARMIJO: Ductility as measured.

MR. CLIFFORD: It would be five percent velocity strain as measured using ring compression test on a double-sided or steam oxidized sample of this size and then you start saying, "Well, okay. If that's it, how do you prepare the sample?" You don't want to get into sample preparation in a rule. You want to move that out and now you start saying, "Okay. Well, it's in a ring compression. What's your rate that you're loading? How much loss are you applying" like there may be aspects of the test that are down in the dirt that you want to include in a reg guide.

MEMBER CORRADINI: I understand, but it's not -- What I guess I'm trying to understand is the complexity is what to put in the rule and to put in the reg guide or the complexity is to put anything other than a straight-up number. I'm trying to understand what makes it complex. Is that where it goes?

MR. CLIFFORD: Right. Correct. It's where it goes.

MEMBER RAY: And also how can you be flexible?

MR. CLIFFORD: Right.

MEMBER RAY: You mentioned there's a big desire to give as much flexibility, to be innovative and to improve as possible and yet without giving away the goal that we have.

MEMBER ARMIJO: I mean it's going to take some work, but I think it's doable. I think there's enough known that I think you can put together. You could say ultimately you want one percent strain capability, classic strain capability.

Now unfortunately it's not like the melting point of a metal. It's a mechanical measurement and sometime you don't get it on a brittle material and different ways of testing will get you a different answer. So that's why he has to say as measured by a certain kind of test and clear how you put that in the rule or in the reg guide.

MEMBER APOSTOLAKIS: Shouldn't the rule just say what you would expect the test to show, a test to show, and then the reg guide --

MEMBER ARMIJO: Yes.

MEMBER APOSTOLAKIS: So what's the --

MEMBER ARMIJO: There may be different -- There's probably different ways to do it. I'm just saying --

MEMBER APOSTOLAKIS: Just say an acceptable way.

MR. CLIFFORD: The reg guide. The reg guide is always optional.

MR. RULAND: The fundamental question we ultimately face in this matter is making sure we pass the wickets set before us by the lawyers. The lawyers are ultimately going to ask a hard question. Okay. How can you tell something is a violation or not and we typically cannot have those requirements in a regulatory guide. They must be in the rule. So what is in the rule by itself must be able to withstand legal scrutiny so we can say, "Okay. If such and such a thing happens, can we demonstrate that it would be a violation or not?" And that is the part of the complexity of drafting this rule is including sufficient detail in the rule, but not so much detail that it then becomes far too prescriptive.

MEMBER CORRADINI: So if I might just make sure I understand that, the wrong way to do is to say one percent plastic strain using your income pressure test with appropriate procedures and the reg guide says what appropriate is because that leaves too much vagueness to determine whether or not there was a violation by or there was an incorrect, inappropriate --

MR. RULAND: That would be one way to think about it, yes.

MEMBER ARMIJO: Approved procedures and then they have to come to you with their preferred test and get your concurrence. That's an acceptable way of doing it compared to what's already in the reg guide.

MR. CLIFFORD: In an ideal world, you would define the test procedures or test protocols in an ASTM standard and reference the standard in a rule. That's the ideal.

MEMBER ARMIJO: That would be nice.

MR. CLIFFORD: But I'm not sure if we could get there.

MEMBER ABDEL-KHALIK: If one were to measure the ductility with a bend test, for example, what would be the corresponding acceptable plastic strain?

MR. CLIFFORD: Do you want to take that one?

MR. MEYER: Yes. I think it would be the same.

MEMBER ABDEL-KHALIK: It would be the same. So why worry that's classifying the method of testing if that's the case?

MR. MEYER: Well, for example, this ductility threshold that we talk about which is shown on this line is actually an intercept of a bunch of ductility measurements of some very low value and you have to decide where zero is.

MEMBER ABDEL-KHALIK: The point I'm trying to make is that all the argument back and forth pertain to the difficulty of being so specific about the method that you're going to use for testing to show that you have one percent plastic strain and if it doesn't matter what method you use you can just specify that you'll have one percent plastic strain.

MEMBER ARMIJO: You have to tie that other method to the database that generated the --

MEMBER ABDEL-KHALIK: But that's why I was asking.

MEMBER ARMIJO: -- these values. And in a way you'd have to say, "Okay. The ring compression test will get us one percent strain and we set our criteria based on that. Now I want to use a tensile test or a bend test." Then you'd have to do make a connection that said that they're equivalent and prove to the staff that they were. Then you probably would get an approval to do that.

CHAIR SHACK: One problem is that you're not actually measuring the plastic strain in this test. You're measuring one percent plastic strain in a ring compression test. If I did the finite element analysis of what the true plastic strain was all through that ring I would get a range of answers.

MEMBER ARMIJO: Right.

CHAIR SHACK: And if I tried to do the same thing with a bend test, I would have to have -- to get the same local plastic strain, I might well have a different criteria.

MEMBER RAY: The embrittlement is very non-uniform. I don't know whether that goes to the bend vs. ring test. But that was one of the points they made.

MR. CLIFFORD: I mean, this is something we're going to have to work through.

(Simultaneous conversation.)

MEMBER ARMIJO: I don't think this committee is going to solve your problem, but it's a real problem. But I think there's ways to get at it and with time I think the staff will work that out.

MR. CLIFFORD: Okay. A few slides on implementing the alternative post-quench ductility criteria. Even if we were to have a single line or curve in the regulation which you could say, "Well, that's really not performance-based" when you convert that allowable CPR versus initial hydrogen as I showed in a previous curve or set of curves taking into account the alloy oxidation cannot and hydrogen pickup, actually it's really hydrogen pickup relation, you are going to get a performance-based family of curves for each alloy.

Now the Optional Test Program would provide flexibility to the licensees where they could go and run tests on their specific alloys. They could run tests to not only define something specific for their alloys, but they could define criteria specific to their transient like, for instance, each size break when you simulate a LOCA, a large break vs. a small break. You could have a different style, a different temperature profile.

Maybe one will have a really fast quench and one will have more of a slow cool. And you could define -- you could run tests to define criteria for here's different transients. Here's a slow cool.

Here's a quench and expand it further where you could say, "Well, what if I want to account for burn-up and the fact that different rods are going to be operating at different powers. Fresh rods are going to oxidize at 2200 degrees Fahrenheit, whereas burnt rods may only get up to 1900 degrees Fahrenheit and we've already shown through some testing that there's some potential benefit to be gained by testing, by doing your ring compression test, on samples that have been oxidized at a lower temperature. So you could have a family occurrence for samples oxidized at 1900 degrees versus these ones were done at 2200 degrees. So you can make this as complex or as simple as you want depending on how much margin you need for your given alloy or for a given plant design.

Path Forward. There are a few ongoing research activities which are being done to enhance the existing technical database. The first one is very important and that is the development and validation of a comprehensive performance-based test procedure for, as I mentioned, the flexibility of running tests to establish post-quench ductility and breakaway oxidation times.

MEMBER ARMIJO: You have two test procedures then.

MR. CLIFFORD: Correct. Absolutely.

MEMBER ABDEL-KHALIK: Could you please go back to the previous slide. If you allow people that level of flexibility, that still does not change the fact that the current rule is non conservative with regard to the 17 percent. Is that correct?

MR. CLIFFORD: That's correct.

MEMBER ABDEL-KHALIK: Okay.

MR. CLIFFORD: Even if we weren't going to add the flexibility, we would have to change the rule because of the 17 percent. It does not always show conservative results or post-quench ductility, I guess.

MEMBER ARMIJO: If you -- There is inherent margin in these materials that haven't been taken into account yet.

MR. CLIFFORD: Right.

MEMBER ARMIJO: And there's inherent margin in the system behavior that hasn't been taken into account yet. But if you made one size fits all, you're going to wind up with a conservative rule that's over conservative for some materials and for some situations and if it turns out to be acceptable to a licensee, they can come in with a justification, topical report or something that would show that they have the adequate ductility by virtue of real test data, using real test data. So I think --

MEMBER ABDEL-KHALIK: But my question was the graphs that he showed earlier showed, for example, ZIRLO crosses the 17 percent line with a burn-up of about 30,000 megawatt-days per ton. If people were to do all these tricks and apply different limits to different LOCAs, different cooling rates, different initial temperatures, etc., that will still be a constraint for some transients.

MR. CLIFFORD: It may for certain alloys, but for certain other alloys we'll actually benefit from this. Other alloys will see that they can go above 17 percent for almost the entire cycle and if they start getting into where they start looking at 19 percent, I'm sorry, you know, oxidizing at a lower temperature for their higher burn-up fuel they're going to buy themselves more flexibility. But, yes, certain alloys that have a high hydrogen uptake are going to be somewhat penalized and they will require more time and effort and more complex reload analysis.

MEMBER ARMIJO: It will encourage the development of better alloys and discourage the use of materials that are marginal and that's a good rule.

(Off the record comments.)

MEMBER BROWN: Sam explained to me the other day. I asked the question about if you're changing your ramp, your linear generic line, just for example, the option that you had in the thing, that the existing fuels you could still use the depleted fuel rods as long as you put them in areas of low power so that they now put under. So the non conservative you try to put them in a high power -- and maximize fuel utilization which is desirable I guess economically. But you don't. You take a hit. You put them in a low power area. Then you still maintain a satisfactory safety margin relative to this phenomena, the hydrogen oxidation, breakaway, whatever.

MR. CLIFFORD: I don't see anything we're doing here is discouraging licensees from going into the spent fuel pool.

MEMBER BROWN: Well, except that they have to know that they have to do that. If you change the rule, they would have to know that they now have to have a better evaluation or knowledge of their fuel conditions so that they make sure the higher burn-ups in low power locations. I wanted to confirm my understanding.

MR. CLIFFORD: That's 100 percent correct.

MEMBER BROWN: Okay. So it's not like the world's falling apart right now. It's just they have to be more careful about how they do it.

MR. CLIFFORD: Right.

MEMBER POWERS: How would they be more careful than you would be in a reload analysis?

MEMBER BROWN: I'm not a reload analysis guy. It just seemed to me that if you put it in a low power part in a reload analysis or wherever you do it.

MEMBER ARMIJO: It would tend to keep you from doing something really foolish to drive a really high burn-up rod just to get the last few neutrons out of it. People won't do that anyway. It's not that economical. The high burn-up fuel just it doesn't make a lot of sense to try and stuff it into the middle of the core. Some of the core designers may like to do that but this would discourage that.

MR. CLIFFORD: And right now, the power envelope for a fuel rod is limited by thermomechanical criteria, mostly rod internal pressure.

MEMBER ARMIJO: Right.

MR. CLIFFORD: And you can't have a rod operating at 60 gigawatt-days operating at 14 kilowatts a foot.

MEMBER ARMIJO: Right.

MR. CLIFFORD: You just can't. First of all, you could never get there and second of all the fission gas release from these would cause the thing to pop. So that's limited now. Would this be more restrictive? I doubt it.

MEMBER ARMIJO: I think it's going in the --

MEMBER BROWN: That's not fair looking for inter TMOL curve relative to the --

MR. CLIFFORD: Yes, they're not related.

MEMBER BROWN: They're not apples and apples.

MR. CLIFFORD: Right.

Okay. We have three ongoing activities as I mentioned. The first one is the comprehensive test program which is very important because we've already seen lab-to-lab variations and results which we don't want to be in the situation where in the future we ask licensees to spend millions of dollars to come up with one test and come up with criteria and then we start questioning the validity of the results. So it's very important to come up with a very comprehensive, detailed test program that the staff finds acceptable that the industry can just follow. The second and third activities are a few more testing done at

Argonne to enhance our understanding and overall complete detectable dose.

MEMBER APOSTOLAKIS: I get the feeling that you are not too enthusiastic.

MEMBER ARMIJO: You have a bad feeling there, George.

MEMBER POWERS: Well, he gets this feeling because there's always a few more tests to do.

MEMBER APOSTOLAKIS: Yes. And it's difficult to this.

MEMBER ARMIJO: You've got to do it right, George.

MR. RULAND: It's our goal to make these few more tests the last few more tests.

MEMBER POWERS: I'm enthused about that, but I'm suspicious.

(Laughter.)

MEMBER ARMIJO: Go on.

MR. CLIFFORD: Okay. This slide just identifies a process we're considering called the Advanced Notice for Proposed Rule-making. The key feature of this process, it allows us to complete the rule-making while, I'm sorry, to complete the few more research tests while we're working with the industry and the public stakeholders to give opinions on the rule and just allows us to work in parallel as opposed to working in series.

And the last slide --

MEMBER POWERS: This Advanced Notice for Proposed Rule-making has traditionally been used when the staff has questions about exactly how a rule ought to be written. What are those questions that are in there?

(Off the record comments.)

MR. CLIFFORD: Well, I think the key feature of the ANPR is going to be asking the industry to comment on the detailed comprehensive test program. To me, that's the biggest issue. Because if we put out a test program and the industry finds out that they can't follow, then it has no use. So that certainly is the key to the ANPR. There's a lot of other questions to ask as far as the impact and implementation and when you actually write rule language ask them to comment on the rule language.

MEMBER ARMIJO: Ambiguity. Clarity.

MEMBER ABDEL-KHALIK: Now the point was made during the subcommittee meeting that for high burn-up fuel there is significant azimuthal variation in the hydrogen pickup.

MR. CLIFFORD: Yes.

MEMBER ABDEL-KHALIK: And if that is the case, do you have any hope of coming up with a test procedure that would give you a definitive answer as to the ductility given the fact that whatever results you're going to get from a ring test will depend on the orientation of the ring?

MR. CLIFFORD: Absolutely.

MEMBER ABDEL-KHALIK: And if you have significant variation in the hydrogen concentration, you can get results that vary significantly. So whatever graphs --

MR. CLIFFORD: One way of addressing that is when --

MR. MEYER: I don't think that's the correct assessment. The ring compression test will not give you good deformation data, good quantitative deformation data, but they are very good at identifying where the transition occurs from ductile to brittle behavior and it does not depend strongly, I don't even thing weakly, on the orientation of the specimen.

MEMBER ABDEL-KHALIK: This just can't be right because the underlying hypothesis here is that the transition depends on the amount of hydrogen.

MR. MEYER: That's correct.

MEMBER ABDEL-KHALIK: And if there is significant variation in the amount of hydrogen --

MR. MEYER: Yes.

MEMBER ABDEL-KHALIK: -- you can't tell me that the transition is independent of how much variation you have in the ring.

MR. MEYER: I'm sorry. No. You're absolutely right regarding the true variation of hydrogen because it will tend to find that place where the hydrogen has peaked and break there. And so if you then go to that fracture location and measure the hydrogen you get the right answer.

MEMBER ARMIJO: Right.

MR. MEYER: And that's exactly what we've done.

MEMBER ARMIJO: Right. So it's kind of self-correcting in that the most embrittled parts of the ring or the sample as where it cracks and that's where the hydrogen is concentrated. That's from your test data.

MR. MEYER: Yes. We've measured hydrogen concentrations azimuthally around the specimen for our six different locations and correlated that with the ductile-brittle transition.

MEMBER POWERS: It might affect your

ability to compute your hydrogen in your actual fuel samples to correlate with your ductility.

MR. MEYER: Certainly. It makes it difficult to develop a correlation for a given fuel cladding type as a function of burn-up because you do have this variation. The variation doesn't seem to show up strongly until you get very high hydrogen concentrations. So at the lower concentrations, I don't think there's -- We haven't seen much.

MEMBER ARMIJO: But it's an issue, but that's the way normal fuel rods work. Depending on the fuel rod design, there will be variability along the length and circumferentially on some types of fuel.

MEMBER ABDEL-KHALIK: But the whole purpose is to come up with a unique design guide that gives us a unique limit that people would apply in their safety analyses.

MEMBER ARMIJO: Right.

MEMBER ABDEL-KHALIK: And if there's a huge error bar on that because of the manner in which the test is done or because of the nature of hydrogen pickup and high burn-up fuel, then there is something fundamentally wrong with the process.

MEMBER ARMIJO: It's really the extent to which the laboratory test. The hydrogen charged specimen reflects the properties of a real fuel rod with nonuniformity because in the hydrogen charged specimen it's going to be uniform. It's done in a furnace.

MR. CLIFFORD: And another way to address it is in the implementation. When you develop a hydrogen model, I mean, if you wanted to be overly conservative I should say, when you develop a model to implement a hydrogen-based rule you would have a licensee get a lot of, you know, high -- fuel rods, go to the hot cell and then measure the variation and maybe take a two signal on that or something.

MEMBER ARMIJO: Right.

MR. CLIFFORD: I mean there are different ways to address the variability in the application of a curve. It's something we need to think about.

MR. MEYER: Yes. I think the whole problem is on that side of the equation. Because on the measurement side, we're able to determine what we have and to correlate that with the threshold that we're looking for.

Besides I just can't resist commenting about a whole thread of discussion here, but there's not that much deviation from this behavior that you see in the dataset taken at Argonne. There's almost no alloy dependence. There's a small dependence if you tested a different temperature or if you tested a different cooling rate. So it's not like you're throwing the flood gates open and you can go out and run a bunch of tests and get wildly different results. They're going to be very close to the results that we're indicating with that straight line.

MEMBER ARMIJO: And I think to your point, Ralph, where you have the most variability in circumferential hydrogen is under high burn-up rods which have the least driving forces from the standpoint of peak temperatures. Yes. It's complicated, but you can handle it. I think you can come up with a practical test that's meaningful.

MR. CLIFFORD: And also if you draw the curve, if you take your empirical data and you draw your acceptance criteria based on the average hydrogen and essentially slid the curve in the conservative direction relative to the peak, it almost penalized the experiment due to these axial or circumferential variations. It's got to be looked into.

Okay. My last slide is just essentially the process for rule-making and it lists some milestones. I'm not going to walk through each, but we're essentially right here step one. As we mentioned the ANPR we're considering and we're going to go out for public comment, probably have another workshop because it's always good to solicit as much information as we can during this process and at some point we'll need to a backfit for 10 CFR 50.109 and then the ACRS will be back here by the time we get to a proposed rule. So you guys will obviously be in the loop.

Any more questions?

MEMBER MAYNARD: On the backfit determination, is it your intent to make this to effect both the existing plants?

MEMBER SIEBER: Sure.

MR. CLIFFORD: That's correct.

MEMBER MAYNARD: Or for existing fuel and stuff out there?

MR. CLIFFORD: No. Our view is that the current rule is inadequate. So we will be revising the current rule.

MEMBER MAYNARD: That's what I thought.

MEMBER APOSTOLAKIS: Is there going to be a regulatory guide somewhere there or after the rule?

MR. CLIFFORD: When we develop Step three, the comprehensive test procedures, we still hadn't decided what the best vehicle is for that, whether it's a reg guide or -- We don't want to commit to where we're going to put it, but it has to go somewhere.

MEMBER CORRADINI: What are your other options other than a reg guide?

MEMBER APOSTOLAKIS: Put everything in the rule.

MEMBER ARMIJO: No.

MR. CLIFFORD: Putting it in the rule would be my last option. I mean that would be my last choice.

MEMBER APOSTOLAKIS: Is somewhere in there you begin to develop the guide?

MR. CLIFFORD: Correct.

MEMBER CORRADINI: Okay.

MEMBER ARMIJO: Okay. Very good. Next is Ken from EPRI.

Let's go.

MR. YUEH: Thank you, Dr. Armijo.

Before I start, I just want to acknowledge the industry's appreciation for collaborative work together with NRC Research Branch and as we move forward to rule-making it looks like things are going to accelerate. I think it's more important to have more exchanges to resolve the many issues.

I have a very quick presentation. The first slide I have is a follow-up to yesterday's discussion of safety significance, you know, what we just talked about. Industry position, hydrogen is a surrogate for irradiation. Talk about data gaps, three areas, estimate of implementation costs and summary.

This slide is actually from Dr. Meyer's presentation on I guess different scenarios where the allowable CP-ECR will fall. On the lefthand side, this chart shows a Westinghouse 3-loop. I think it's a core-to-core model. It has a peak from each of the sampling. It basically just shows core cycle, 2nd cycle and third cycle.

What I tried to do is plot ECR for a typical LOCA scenario onto this chart. The first cycle -- Basically I stretched the temperature at the pickup location to 1200 degrees Celsius and then calculated the ECR. For a realistic scenario, the ECR is way down here, you know, in the first cycle, the second cycle and third cycle.

And then I did the maximum. I assumed the sample was, the fuel rod was, at 1200 degrees Celsius and then I calculated the time where you would reach that peak and then I scaled down based on the difference here that you can expect in the first and second cycle. The ECR, the achievable ECR, is scaled below the potential New RIL-0801 and taken together with potential pickup cladding temperature which is shown in the red line which Dr. Meyer asked if it could be done at four percent. I added three percent. This is where this line falls.

I do want to point out also that this line here IN 98-29 I think is based on a hydrogen pickup fraction of about 25 percent. I think that's a abnormally high pickup fraction. I think the average of Zirc-4 is about 15 percent. If 15 percent pickup fraction is used for this line it will fall somewhere along here. So clearly from this there is no safety concern.

MEMBER ARMIJO: That may not be clear to everybody what you just said.

MR. YUEH: There may be odd core designs as you discussed earlier where somebody could have a high -- fuel in the middle, but those would be exceptions. I think in general 90 percent, 95 percent, of fuel samples will behave this way.

MEMBER ARMIJO: So your expectation is Zircaloy-4, modern zircaloy-4.

MR. YUEH: Yes.

MEMBER ARMIJO: Could meet these new criteria.

MR. YUEH: Yes.

MEMBER ARMIJO: Does. That the criteria -- line that you're in trouble.

MR. YUEH: I do want to follow up on what was discussed on the hydrogen level on the ring test.

MEMBER ABDEL-KHALIK: Before we go there, this must be a very conservatively designed core because the maximum pin power is 1.4.

MEMBER ARMIJO: Those are peaking factors.

MEMBER ABDEL-KHALIK: Right, but what is the -- That means the bundle average power peak value much be way below 1.4.

MR. YUEH: It's a little bit. This is a peak rod.

MEMBER ABDEL-KHALIK: Yes, I know. This is the peak rod.

MR. YUEH: Yes.

MEMBER ABDEL-KHALIK: Which means that the peak bundle average power must be way below 1.4.

MR. YUEH: Yes.

MEMBER ABDEL-KHALIK: Which means that this is really a very conservatively designed core.

MR. YUEH: Yes. So this kind of power --

MEMBER ABDEL-KHALIK: It's sort of biased story, don't you think?

MR. YUEH: Sorry.

MEMBER ABDEL-KHALIK: This would be a biased story. That's the point you're trying to make.

MR. YUEH: But I do want to present a case from the high temperature side to show that it still falls below the line, the new one.

MEMBER ABDEL-KHALIK: Okay.

MR. YUEH: That's the point I want to try to make even for a very conservative core design where you have high power and then I scale the temperature to the maximum 1200 Celsius. If the second cycle is limited here, it's still below the --

MEMBER MAYNARD: You're saying that this shows that they're safety significant. Aren't you also showing that there would be no problem meeting the new rule?

MEMBER SIEBER: That's right.

MEMBER ARMIJO: For that particular design.

MR. YUEH: Yes. Well, for -- That's why we're trying -- On Tuesday, Westinghouse was trying to argue there is no real benefit because the new limitations from hydrogen, it really applies to high burn-up fuel and with high burn-up fuel the temperature is much lower, less than 1000 Celsius. You accumulate almost no ECR and ECR accumulation is only significant above 1000 Celsius and below that it's very slow and even there's a big difference 1200 and 1100.

And the point that we just talked about with high burn-up fuel rods, even though you have a lot of hydrogen in there, the temperature is not capable of reaching temperatures. You cannot accumulate ECR. So the ductility there is not an issue because --

MEMBER ARMIJO: But it does point out that the real risk, the safety risk, is in your second cycle or first cycle fuel that has plenty -- that's had some burn-up and if you pick up too much hydrogen you're going to get in trouble.

MR. YUEH: Yes. I have used 400 ppm hydrogen for end of the second cycle and I've been criticized that that's too high for other industries. It's too high because Zirc-4 it is a little bit on the high side and for advanced alloys these datapoints were even below it.

MEMBER SIEBER: But there's some

subtleties in here that can change this a little bit, too, for example, addition of burnable poisons and how they're lumped and the extent to which they burn off over -- The total burn-up of the fuel can change these power ratios.

MR. YUEH: There could be temporary spikes, yes.

MEMBER SIEBER: Well, it could be more than slightly depending on how aggressive you tried to do these things. The fact though is that the current rule is technically not correct.

MR. YUEH: Right.

MEMBER SIEBER: And you can meet the new rule doing the same fuel design and manufacture that you do now. So you know your argument is to make the rule technically correct and not disturb fuel design, fuel manufacturing, core -- The core analysis seems to me not all that bad a deal to just to achieve more technically a correct picture by rule as to what's going on in the plant.

MR. YUEH: Yes, exactly. It's going to coast $300 million to do.

(Simultaneous discussion.)

MEMBER ARMIJO: Go see the Treasury; they're giving money away.

(Laughter.)

Okay. We really have to move along.

MR. YUEH: I will just say this. This here is the test data for red irradiated and black hydrogen pre-charged. I just want to show that the data falls on the same trail line even though the tubing material is made from different manufacturers and different processes and they all fall on the same line and the industry certainly believes that hydrogen can be a surrogate for irradiation.

CHAIR SHACK: Are you planning any more work to support that?

MR. YUEH: Apparently, NRC Research is planning to conduct tests to fill up a couple of datapoints in here.

MEMBER ARMIJO: The answer, yesterday, at the subcommittee meeting, they presented data saying they were going to do more.

MR. YUEH: Yes.

MEMBER ARMIJO: At EPRI.

MR. YUEH: We are going to do non-irradiated.

MEMBER ARMIJO: I understand the non-irradiated stuff.

MR. YUEH: Yes.

MR. MEYER: What the NRC is going to do is we're going to get one more red point in the middle of that.

MEMBER ARMIJO: Will that be on ZIRLO?

MR. MEYER: Yes.

CHAIR SHACK: I guess the question is do you presume this is not true for all zirconium-based alloys or is that something that has to be demonstrated?

MR. MEYER: If you're asking me, the answer is I'm quite sure it's true for all zirconium-based alloys based not only on the data that we've taken which you see is quite consistent but also on our understanding of the process that causes this embrittlement which doesn't look like it should be strongly dependent upon the alloy.

CHAIR SHACK: That certainly makes people very happy.

MEMBER CORRADINI: Yes, the vendors.

MR. MEYER: But you have to make a clear distinction between this and the breakaway process because -- And I'm not sure that that distinction has been made clearly here because the breakaway process which is a totally different phenomenon is sensitive to alloy and fabrication details.

MR. CLIFFORD: I think it's important to notice I don't think the staff has consensus on that issue. I would like to see in my opinion some of the irradiated tests and there are many irradiated tests up there. I mean, each one of those points is a handful of tests, not just a single one. I would like to see the same stock material pre-hydrided and run for a direct, you know, if you have a irradiated piece with 425 ppm, I'd like to see the same stock material pre-hydrided to 425.

CHAIR SHACK: You want to see the true comparison.

MR. CLIFFORD: Into a true comparison and see where it falls with the exact same point. So on my slide Tuesday, I believe I had a bullet that said it's up to the industry to validate the pre-hydrided.

CHAIR SHACK: I sort of noticed it was missing from your slide today.

MR. CLIFFORD: There was a lot of material missing from my slide today.

CHAIR SHACK: All right.

MR. YUEH: That's something we can follow up on if they can find stock material. We don't know if the material is there.

MEMBER ARMIJO: Okay. We should go on.

Keep moving.

MEMBER POWERS: Can I just ask?

MEMBER ARMIJO: Yes, sir.

MEMBER POWERS: About this concern about the region between 100 and 300 ppm? Is there someone out there that thinks there's a resonance or something like that that they were going to see something wildly deviating from this line?

MR. YUEH: Well, you do have the non-irradiated datapoints there and in a LOCA all the irradiation history, it's mostly wiped out.

MEMBER POWERS: What I'm asking, why is it so crucially important to get a datapoint? I mean, you got -- I have no idea whether this is a 100 to 300 datapoint occurred or a straight line through these datapoints. But unless somebody has some hypothesis that there is some sort of a resonance phenomena occurring between 100 and 300 datapoint, it's difficult for me to imagine that it's crucial to get a datapoint in there.

CHAIR SHACK: Which I think is why it's the last test Ralph is running.

MEMBER POWERS: If I were going to hit datapoints out here and add to this, it would be in the 600 to 700 range.

MEMBER ARMIJO: Let me point that there is two zircaloy-2 datapoints on basically fresh cladding on this curve, a whole bunch of ZIRC-4, ZIRLO C5 and I'm just not a full believer yet that this one curve is representative of all alloys. In a laboratory test, many you can do that with hydrogen charging, but it doesn't necessarily correlate with irradiated fuel.

I think it's conservative. I believe it would be conservative, but I just think we're putting apples and oranges and we're saying that they're all going to behave the same and probably in a laboratory test of unirradiated material that you hydrogen-charged, you're going to get very similar results because they're basically zirconium. But in a fuel rod, that's not evidence to me anyway. The irradiated datapoints in that middle range are needed. That's what I think.

MEMBER POWERS: But then it's going to be wildly off to the right or the left. How far could it go?

MEMBER ARMIJO: If you have a big gap. Look here. You have an irradiated datapoint at 100 ppm hydrogen and at 550. You have a big gap in the middle and you fill it in with --

CHAIR SHACK: The tests will be run soon.

You can buy the loser a beer.

MEMBER ARMIJO: I'll buy him a beer anyway.

(Laughter.)

Anyway, they're going to do the test. At least, that's what I heard.

MR. RULAND: For the record, the staff has had this same debate.

(Laughter.)

MEMBER ARMIJO: Okay.

MR. YUEH: Some gaps in data space and industry concerns. Those were worked on at ANL at 1200 Celsius and the benefits to conduct testing at lower temperature. You get higher ductility for the same ECR. This is where industry is conducting testing at 1200 degrees as well as the lower temperatures with different hydrogen pre-charge.

We were concerned with requirements for 2-sided oxidation away from the ballooned region. It's right now not supported by ANL data at Limerick. The integral tests does not show a very significant alpha layer build-up on ID.

And then with respect to the Halden test which is a reference in the RIL, the impact of an ID oxide, the oxygen diffusion into the material that burns up the cladding, I think still needs to be quantified. And with this uncertainty, at this point it's difficult for me to imagine how this could be incorporated into the rule. Once it's in the rule, it's very difficult to change and then the industry feels that that should be in a lower level document.

The periodic testing on breakaway oxidation at this point is driven by the short E 110 breakaway time. Paul Clifford has already stated that the shortest breakaway time for the Westinghouse alloys is 3,000 seconds and the system, the procedure, response before that. So that sort of refutes the need for periodic testing or treat this as a concern.

Also by the time you reach 3,000 seconds at 1,000 Celsius if the 2-sided oxidation is enforced, you will have also reached the 17 percent you see at 1,000 Celsius. So these are sort of arguments against it.

The industry believes that breakaway oxidation can be addressed through a QA program. The short breakaway time absorbed in E 110 is due to the initial processing of the material. The electrolytic process from the beginning uses fluorine in the process that fundamentally changes the process and nobody in the west is using that process. I think even the Russians themselves some of them are away from that.

Implications for Proposed Change, Cost Estimate, you know, the proposed change will require a re-licensing. All operating reactors will need to demonstrate compliance. The vendors will also need to conduct expanded hot cells to measure hydrogen to be able to resolve the correlation of -- between corrosion of hydrogen because hydrogen is not readily measurable. Oxide is.

The costs to vendors and licensees to comply with the anticipated new rule is estimated at several hundred million dollars. And in addition to that, implementation will be really resource intensive. Each application will need to be reviewed and somebody estimated that in years for the efforts. And if this proceeds, the industry will obviously request a phased in implementation.

MEMBER ABDEL-KHALIK: No, part of that is sort of the explanation for the several hundred million dollar figure was the comment that they would have to redo the LOCA analysis for each reload to see if it complies with the new rule.

MR. YUEH: Also, each plant, I think they have re-license.

MEMBER ABDEL-KHALIK: Right. But don't they already have to redo the LOCA analysis for each? I don't understand that.

MEMBER ARMIJO: Why wouldn't you just do it with a --

MEMBER SIEBER: They don't do it that way.

MEMBER ARMIJO: They don't. For new material, you could come in with a licensing --

MEMBER SIEBER: No, they don't.

MEMBER ARMIJO: I'm talking about new material.

MEMBER SIEBER: Reload safety analysis can be done by seeing if your important core parameters fit in the box of the outline of the window that the LOCA analysis covers which you means you don't have to redo the LOCA analysis. You have to make a parameter comparison between the limits of the bounding LOCA analysis to the core parameters of the reload you want to install which is a relatively quick and relatively inexperienced process.

In order to implement a new rule as I picture it, each vendor is going to have to come up with a new set of core analysis done from scratch to rebuild the envelope upon which reload analysis must fit and it's been awhile since I actually signed the bills to pay for this stuff, but I don't remember the cost of that.

MEMBER ARMIJO: It's certainly going to cost something, but I can't see several hundred million dollars. I think that was an estimate when there was a vision that the staff was totally inflexible and was going to ram some arbitrary overly conservative criteria.

MR. YUEH: I was surprised about the numbers myself. You know, Westinghouse did the estimate and we had phone calls with utilities and they agree with the numbers. I'm not an expert in this area. So I don't really know, but that's what they --

MEMBER ARMIJO: Okay.

MEMBER SIEBER: Try to picture if you did an actual core analysis for every plant as opposed to the enveloping technique on every reload where you could build up the bill.

MR. YUEH: This is for every -- for the every plant.

MEMBER MAYNARD: If there are 100 plants and you have to do LOCA analysis and especially if there's any retesting that has to be done.

MEMBER SIEBER: The testing may drive the cost.

MEMBER RAY: It seems to me like the industry ought to be thankful that nobody is running around here saying the sky is falling and that we're persuaded by the reasonableness that has been described here today that this is something that can take place on the kind of timeline that is described. But it just doesn't seem to me plausible you can walk away from it and do nothing and say this must not be a problem. I mean, that's ridiculous.

MEMBER SIEBER: the current rule in my view is technically an error and there is new phenomenon that needs some further investigation.

MEMBER RAY: Yes.

MEMBER SIEBER: It may apply. It may not apply. You can't just say it doesn't exist.

MEMBER ARMIJO: The expanded hot-cell campaigns to license corrosion hydrogen correlations, every manufacturer has a database of hydrogen corrosion going back to time and memorial which they use in a variety of ways and they're going to continue to gather that data anyway. For new material, you'd have to generate that data and you should. You shouldn't be introducing a new material without sufficient basis.

And it's expensive I grant you. For new materials this would be expensive. But you would do it anyway to introduce some cladding.

MEMBER MAYNARD: Now you can get the cost of --

MEMBER ARMIJO: Yes.

MEMBER MAYNARD: First of all, I think that the worst case estimate with unknowns of what's going to come out. The other thing it's total industry. It's not a per plant basis. So I think we need to focus more on what's wrong with the rule and we need to -- But I think that analysis will sort that out once a rule is kind of drafted out.

MEMBER ARMIJO: Okay. You have a summary.

MR. YUEH: Last slide. The evaluation so far indicates there are no safety concerns and the industry does not see a need to rush through the rulemaking. The industry supports a flexibility rule. There's some elements of it already proposed and proposing or using lower level documents for details. There are regulations guides.

The industry supports qualification testing, but not rule-mandated periodic testing and the final comment we have is pre-hydriding appears to be a good surrogate for irradiation. Thank you.

MEMBER ARMIJO: Okay. Questions?

MEMBER POWERS: Yes. I'm still perplexed by your comments on periodic testing. It may be because I admit to a certain failure to understand the subtleties of breakaway oxidation. It has always been my interpretation of the E 110 experiments is that it simply said, "Gosh, these always can be very sensitive to things that you don't anticipate readily."

And that the concept of periodic testing came about because there are things that change in processes that you don't know about. I think Dr. Shack brought up the possibility that changing the solvent you use to wipe down the cladding could induce an impurity that subsequently affected breakaway oxidation.

All that seems very reasonable to me. Why shouldn't you have in the program a periodic test for breakaway oxidation to make sure these uncontrolled and unknown things aren't creeping in some way either in your process or in your supplier's processes.

MR. YUEH: At this point, what caused E 110 is really not known and we think it's related to the electrolytic process and obviously you can have other things that get introduced into it if you make a process change. But every time you make a process change the process needs to be qualified and we're not saying that don't do it altogether. Whenever you qualify the process, do it at that time.

There are implications from periodic testing. How often do you do it and if you do it if one failed, what do you -- Depending how the gap is you could already have fuel in your reactor and how do you disposition that. And also --

MEMBER POWERS: So what you're saying is that there is between each conscious change in a process there are absolutely, positively guaranteed and assured no changes occur. Even though they earn emission, you don't know exactly what change precipitated the problem in E 110.

MR. YUEH: We think that the E 110, we don't know exactly what it is, the missing element or extra elements in there, but we believe it's linked to the electrolytic process which Westinghouse is not using.

MEMBER POWERS: This is a faith-based experiment.

MEMBER ARMIJO: No, it's a flag-based, Dana. I have to step in and --

MEMBER POWERS: No, you don't step in because I'm talking right now.

MEMBER ARMIJO: Okay. I'll step in after your opinion.

MEMBER POWERS: After my opinion, all right. I'm asking is that your hypothesis. Do you assure that there's absolutely and no possibility of an unknown change occurring in a process between conscious changes?

MR. YUEH: There's no -- From my experience, there are no absolute guarantees. We look at what is practical, you know, realistic.

MEMBER ARMIJO: Can I step in now?

MEMBER POWERS: You may.

MEMBER ARMIJO: Ken may not have run a fuel factory, but I have and cladding is one of the -- Other than the pellet, cladding is the most important component in a nuclear fuel fabrication and enormous care is taken into every step in that process and we've learned a lot over the years. There is periodic testing going on in the laboratories for just normal corrosion behavior so something doesn't go wrong.

You don't change cleaning solvents willy-nilly. You don't change heat treatments willy-nilly. You don't change belt polishing. All of these things are under very tight quality assurance controls. People can screw up data. I don't deny it.

But in my experience and what I've seen, I've never seen a failure of a fuel rod by corrosion, variable corrosion, unanticipated high corrosion, never heard of in PWRs and I've never seen it in BWRs for which I know a lot about. So corrosion variability is really kind of more of a visual and a problem, but it's not a failure problem because of these tight controls.

E 110 was a bizarre material. It's made by a totally different process starting with the zirconium that they make by an electrolytic process which is not the same as pro process we use and it's not as clean.

The other thing is the material was cleaned by some perhaps fluoride containment result and the ANL data shows that if you remove that surface you get much better performance on this susceptible material.

And the third thing is in the testing that's been done on breakaway corrosion by ANL they've demonstrated that there's a lot of margin to the limit, let's say, for small break LOCA. But if you were right on the borderline your material would go into breakaway corrosion in 1500 seconds and your time was 1400 seconds. Sure. Maybe you ought to change materials completely. Maybe you shouldn't be doing periodic testing. So it depends on how robust and how much margin you have to the allowable time.

All of these things, it boggles my mind to say because of a goofy Russian material, my apologies to Mr. Putin, because of that that American fuel manufacturers have to go in through a periodic regulatory, mandated testing process is just not -- It just doesn't make sense. Whether they do it in their factory is part of a normal thing. It's another thing to worry about. That makes sense. But then getting into it, turning it into a regulatory number for regulatory staff and manufacturing staff are circling around this nonsensical process just strikes me as something we ought to avoid.

MR. YUEH: I also want to add that this phenomena only happens here at the material at a fixed temperature for a period of time. In a real LOCA, the fuel rod is not sitting at a single temperature. It's going to go through a cycle and whether you still see breakaway even after an extremely long time is still in question that it may not happen if you just pass it through it.

MEMBER ABDEL-KHALIK: Can I just ask the question, a follow-up, to what Sam said? Were runaway oxidation tests routinely done as a part of alloy development?

MEMBER ARMIJO: Alloy development, sure. All sorts of bizarre tests are done in alloy development.

MEMBER ABDEL-KHALIK: So this phenomenon was known and recognized as being a non-issue for the alloys that -- Or was it just by the grace of God that we use processes that didn't result in this issue?

MEMBER ARMIJO: I think a little bit of what you're saying that the materials that we developed for good corrosion resistance tended to be pretty good. Our data shows that they tend to be pretty good for breakaway corrosion as well. But nobody just routinely runs zircoloid cladding at 1,000 degrees or 900 degrees Centigrade for 5,000 seconds or 6,000 unless there's some new phenomena that they're worried about and that's what the breakaway corrosion tests are supposed to do.

MEMBER SIEBER: Breakaway corrosion is sort of a unique --

MEMBER ARMIJO: It's new. It's not your standard thing.

MEMBER SIEBER: It's recently been discovered and the kind of testing program you would do in the development of an alloy probably would not identify it. On the other hand, it appears to be, at least to me because I lack knowledge, concentrated in one kind of an alloy.

MEMBER ARMIJO: Yes, if you were going to make an alloy today and you didn't do a breakaway corrosion as part of that qualification that would be irresponsible. But if you already know that the cladding you're making today with standard, very controlled processes has a lot of margin against breakaway corrosion based on testing, then why in the world do you mandate some regulatory requirement that it has to be repeated every so often because your factory is out of control or could be out of control.

MEMBER SIEBER: Well, the PAKS failure in Bulgaria, that fuel sat at high temperature for days before it failed and that was E 110 with --

MEMBER ARMIJO: Well, that's another thing.

MEMBER SIEBER: -- breakaway oxidation. So it's not the kind of thing you would experience during a LOCA accident. It would not sit there and just percolate forever.

MEMBER ARMIJO: Okay. I'm sure there's going to be more debate on this subject. Thank you, Ken, and I'll be quiet.

I'll address the Committee. I think a good issue has come up. The staff has not requested a letter, but some members of the Committee think a letter from the full Committee would be in order. I'm one of them. But I'd like to get a sense from --

CHAIR SHACK: Let's discuss that later this afternoon when we get into our letter writing session.

MEMBER ARMIJO: Okay.

MEMBER SIEBER: It will give us something to do on Saturday.

(Simultaneous speaking.)

CHAIR SHACK: At the moment, I'd like to take a 15 minute break and come back at 3:10 p.m. Off the record.

(Whereupon, a short recess was taken.)

CHAIR SHACK: We can come back into session. Our next topic is the presentation on the NRC Staff's initial White Paper on containment overpressure credit issue, and Mario will be leading us through this discussion.

VICE CHAIR BONACA: We have the connection on the -

CHAIR SHACK: Yes. As far as we know,

that's open.

VICE CHAIR BONACA: All right. Good.

CHAIR SHACK: I should mention that TVA has asked to call in, and is listening in. But, again, they're in listen-in mode, so there'll be no interruption in the presentation.

VICE CHAIR BONACA: Okay. Before we begin the presentation and discussion on the NRC Staff's Position Paper on Containment Accident Pressure Credit, I'd like to make some introductory remarks regarding the ACRS concern on this issue. It will be very brief, because you all have heard those worries, and we expressed them to the Commission a month ago, and pretty much are the same.

Historically, most plants in the U.S. have been licensed with no credit for containment overpressure. This approach preserved independence of barriers to release of fission product, and provided significant margin for the ECCS pumps. Credit for accident pressure was first broadly allowed to licensees in responding to the BWR suction strainer clogging issue. This is an excellent example of how a regulatory margin that is there to deal with what we don't know can be used to address a newly discovered issue. And there are still potential issues that we don't know of that may need some additional margin in NPSH.

Now, remaining margin is being used by some licensees to support voluntary licensing actions, such as EPUs. ACRS has consistently expressed concern with this use of NPSH margin. We have accepted this use of margin if supported by Reg Guide 1.74-type of demonstration of low risk, or if supported by amount and length of credit being small.

Conversely, the Staff approach focuses on regulatory criteria, and imposes no limits on the amount and length of credit as long as these regulatory criteria are met. Margins to material are significantly reduced, in fact, pump cavitation and operator intervention to manage cavitation are allowed under certain circumstances. We are working with the Staff to see that we can resolve our difference on the evaluation criteria, and to determine what information can be requested of licensees that would allow the ACRS to perform its assessment against its own criteria, and these remain different from those of the Staff.

The first step that we have in this process of communication is the White Paper that was put together by the authors which are in front of us, and we haven't had a chance to review it, but it's a significant document, and we really believe that we will need probably another meeting after this to further clarify some of the issues, and would depend on that, or make a decision after we've gone through this preliminary meeting here that I believe is scheduled for two hours.

CHAIR SHACK: Yes.

VICE CHAIR BONACA: With that, I will now proceed with the meeting, and I call upon Mr. Ruland of NRR to start the meeting.

MR. RULAND: Thank you, Dr. Bonaca. Good afternoon. My name is Bill Ruland, and I'm the Director of the Division of Safety Systems in NRR.

On November 4th, we transmitted a White Paper to the ACRS on a subject that the Staff has discussed with the Committee for several years, the use of containment accident pressure in determining the available net positive suction head for ECCS pumps. The White Paper collects our reasoning for this practice in one document, with the hope that it would increase the clarity of our position, and improve communications, particularly with the Committee.

The White Paper is an initial version, and the purpose of the White Paper for us, in particular, was to focus our attention on what our collective position was on this matter.

As we will discuss shortly, we have -- because of that focus, have already identified some minor improvements that need to be done to the document since it was issued. We are trying to emphasize that we are still listening.

Previous ACRS comments have been incorporated into our reviews of containment accident overpressure. ACRS has recommended that we look at the issue from a broader perspective than just LOCA, and we're doing that. ACRS was interested in our concept of applying statistical analysis on this issue. We are now reviewing a BWR Owner's Group Topical Report prepared at the suggestion of the Staff, which applies statistical methods to this issue.

The last two extended power uprates we proposed using containment accident pressure, including realistic calculations in their submittals or in responses to Staff RAIs. We know the Committee is interested in seeing more realistic analyses. We look forward to your comments on the White Paper, and we will revise the paper, if warranted.

We will continue to drive this issue to closure. We believe our position is well supported, as documented in the White Paper, and as we will discuss here today, it is consistent with the assumptions of containment integrity made in other areas of reactor safety.

Finally, we understand that this is a information briefing, and, accordingly, it's our understanding that the Committee does not intend to send a letter based on this briefing. The Staff also understands that based on a recent ACRS Commission meeting, that the Commissioners are considering issuing a Staff Requirements Memorandum in this area. Based on the previous Staff SRM for a Commission meeting, the Staff was requested to issue a Commission paper to the Commission on this issue, basically, if our disagreements continue. So one of the things that we are trying to get out of this meeting is to try to understand how the Committee sees this issue based on us trying to document all this information in one place, to try to make a determination when and if a Staff Commission paper needs to go forward.

With that, those are my opening remarks. If there are no questions, I will turn give the mic to Rich Lobel.

MR. LOBEL: Good afternoon. My name is Richard Lobel. I'm a Senior Reactor Systems Engineer in the Office of Nuclear Reactor Regulation. With me is Marty Stutzke, Senior Technical Advisor for PRA Technologies in the Office of Nuclear Regulatory Research.

The purpose we're here today to discuss the NRC Staff position on the use of accident pressure in determining the available net positive suction head, NPSH, of the ECCS and containment heat removal pumps. In particular, the Staff position and discussion provided to the ACRS in a memorandum to the ACRS Executive Director dated November 4th, 2008. The White Paper enclosed with the memorandum is the result of a Staff re-examination of this issue.

These are the topics we'd like to cover today; the regulatory background provides a perspective on this issue. Dr. Bonaca briefly mentioned some of this. The Staff position has changed over time. I'll discuss the regulatory guidance and policies next, the technical bases for using containment pressure in determining available NPSH. Marty will discuss the risk aspects, and finally some discussion of future actions and conclusions.

MEMBER APOSTOLAKIS: I have a question. Reading the White Paper, I didn't see a clear statement what the failure criteria is. It starts out with an inequality, and the available NPSH must be greater than the required NPSH. Then there is discussion on erosion rates, there's discussion of time, some tests that were done and for half an hour required NPSH was exceeded and nothing happened. Is there a place, or are you going to tell us what the failure criterion is? How does time come into this? How do other things come into this? There are discussions of various aspects, but they don't seem to come to a single inequality or equality that says if this happens, then I have a failure.

MR. LOBEL: Well, that's a good comment. I've been thinking about that, and I think I have an answer. And there's a slide later on that talks about NPSH margin, and I think that's the place to talk about it.

MEMBER APOSTOLAKIS: Okay.

MR. LOBEL: It's kind of an interesting and complicated subject, and I'm not sure that the pump industry agrees on exactly what margin -- well, I know they don't agree on exactly what margin there should be. But I think the basic answer is it's the required NPSH, however that's defined. And as you've seen in the different reviews we've done, licensees have proposed different values.

The Hydraulic Institute has a value and a way of deriving the value, but people in the industry have used different values, and people in the industry have credited for limited amounts of time exceeding the required NPSH, and the pumps in the analysis have cavitated for some length of time. And the purpose of that table with the different tests was just to show that there has been prototypical testing done for that.

MEMBER APOSTOLAKIS: I understand that.

MR. LOBEL: But I think the basic answer -

MEMBER APOSTOLAKIS: If you'll cover it later, that's fine.

MR. LOBEL: Okay.

MEMBER APOSTOLAKIS: I'll wait.

MR. LOBEL: Okay. There are several changes to the White Paper since it was sent to you. First, the White Paper stated that Reg Guide 1.1 would be withdrawn. Reg Guide 1.1 was the NRC's first Reg Guide, and has only a single position, and that is that containment accident pressure should not be used when determining available NPSH.

We've determined that withdrawal is not necessary. Reg Guide 1.1 will be revised to state that Reg Guide 1.82 is the current guidance. Reg Guide 1.1 is part of a licensing basis for some licensees, and it may continue to be used for that purpose. In other words, there's no reason for licensees to go back and change their licensing basis, especially because that's the most conservative position to begin with.

VICE CHAIR BONACA: Are you talking about 1.82?

MR. LOBEL: I'm sorry?

VICE CHAIR BONACA: Reg Guide 1.82 is very focused on the strainers issue.

MR. LOBEL: Well, what we're trying to do with Reg Guide 1.82 is make it the one place that will have Staff guidance on pump suction issues.

VICE CHAIR BONACA: Okay.

MR. LOBEL: So it'll have the guidance on debris, it'll have the guidance on air entrainment, sump design, and NPSH all in one place, so it's going to serve several purposes. A lot of that information is already in Reg Guide 1.82.

MEMBER BANERJEE: Will it be revised?

Because at the moment it says that containment overpressure should be minimized. And there are several places where it addresses this issue, which -

MR. LOBEL: I'm going to talk about that a little later, too. Well, we tried to revise it once, and came to ACRS, and got a letter saying make some changes before you issue it. So my thinking was before we try to rewrite the Reg Guide, we, the Staff and ACRS, ought to come to some kind of agreement, or there's no point coming back with another Reg Guide until we all agree.

MEMBER BANERJEE: But the intent is to revise it.

MR. LOBEL: The intent is to revise it, yes.

MEMBER BANERJEE: Okay.

MR. LOBEL: The White Paper described a process for the risk analysis that had elements of Standard Review Plan 19.2 Appendix D, and elements that were not part of that procedure. We've decided that it's more appropriate to use the existing guidance in 19.2 Appendix D. Marty will discuss this.

The discussion of NPSH margin needed to be revised, and I will do that -

MEMBER ABDEL-KHALIK: With regard to the

second point, is there a clear and distinguished equivalent statement as far as EPU submittals?

MR. LOBEL: For what?

MEMBER ABDEL-KHALIK: I mean, you're saying for non-EPU submittals risk procedures will follow SRP 19.2 Appendix D.

MR. LOBEL: Oh. That's because for EPU submittals, the licensees already provide a risk analysis. What we're talking about is really non-EPU submittals that deal with the subject, since EPU submittals are already taken care of.

MEMBER ABDEL-KHALIK: Okay.

CHAIR SHACK: Just on that point, your summary makes that statement that the 1.82 would be revised to include that request for risk information on the EPU submittals, but the Executive Summary doesn't, so that you have to go all the way to the very end of the White Paper to find that. And it seems to me that's an important enough element, it should go in the Executive Summary.

MR. LOBEL: Well, Marty is going to talk about that. And the position has changed a little, and we want to spend a lot of time talking about that. Well, the reason basically was we were outside of procedure, and we have a written procedure. It's gone through an approval process with our management, and I believe even up to the Commission level, so we want to stick with that process. So it's different in some ways than what's described in the paper, but it's in perfect agreement with other regulatory documents that are already issued. We'll go into all that.

MEMBER CORRADINI: Just to clarify, Said, just so I'm on the same page, so Said's point was what's the converse of the second bullet? And your point is that the converse of the second bullet is that any EPU submittal must have a risk analysis. Following that 19 point, for this purpose. That's what I thought -

MR. STUTZKE: For EPUs they also follow Appendix D.

MEMBER CORRADINI: Okay. That was my question.

MR. STUTZKE: They're non-risk-informed, everything that's non-risk-informed license amendment falls under that appendix.

MEMBER CORRADINI: Okay.

MR. LOBEL: Okay. And we changed the discussion a little bit on NPSH margin, and I'll talk about that.

Okay. ECCS and containment heat removal pumps in BWRs and PWRs are centrifugal pumps. Centrifugal pumps are capable of operation over a wide range of flow rates and pressures. Their operation is well understood. There are numerous books and standards on their use. They're used in a wide variety of applications, but one of their drawbacks is that they're subject to cavitation, as are other devices subject to flowing liquids. Pump cavitation can lead to delivery of less than expected flow and discharge pressure. It can also lead to pump damage.

In some operating reactors, the Staff has allowed use of containment accident pressure and available NPSH calculations to avoid pump cavitation or limit it to a short time at an acceptable level.

Regulations allow use of containment accident pressure in determining the available NPSH of safety-related pumps. The regulatory guidance on this issue has changed over time, as discussed in the White Paper. The Staff has allowed use of containment accident pressure in demonstrating adequate NPSH in situations where calculations predict that the pumps would otherwise cavitate.

MEMBER RAY: Wait a minute. Is there any assumption about what is required to produce this containment accident pressure? In other words, does it matter?

MR. STUTZKE: Nothing special is done in the way of containment accident pressure for these types of calculations except to minimize it. The pressure is predicted to be there as a result of the accident.

MEMBER RAY: Well, supposing it requires the operator to take some action. Is that a consequence of the accident?

MR. STUTZKE: The analysis -- the only operator action that's -- there's a couple of operator actions that are assumed. For a BWR at ten minutes, it's assumed that the operator -- let me back up for a second.

MR. RULAND: You're going to cover this, aren't you, Rich, this whole issue of operator action later, or do you want to answer it now?

MR. LOBEL: Not that question.

MR. RULAND: Okay.

MR. LOBEL: The RHR system in a BWR has several modes. One of them is -- and the way it's set up at first is for injection, so given a LOCA, the RHR system is in the injection mode. At ten minutes, it's assumed that the operator changes the RHR, assuming he has proper core cooling. He changes from the injection mode to the suppression pool cooling mode.

The only other operator action -- and that mode can be either cooling the suppression pool directly, or he can initiate spray. He can also, if he has both trains, he can continue to inject with one train, and use one train for suppression pool cooling, or he can use that train in the spray mode. Either mode cools the containment, because the flow is going through the same heat exchanger, so either way is cooling the containment.

The operator has a caution in the EOPs that if the pressure gets below a certain value, the operator terminates the spray. And that value where no credit is taken for containment pressure is zero psig. If that plan is taking credit for containment pressure, then it would be a different pressure.

The operator in a BWR also has curves, and I was going to talk about this a little later, of suppression pool temperature versus pump flow with pressure as a parameter. So pressure is already considered in the EOPS, and the operator can tell by looking at those curves whether he has adequate NPSH.

MEMBER RAY: So I think your answer is yes, operator action to provide, or insure, or maintain containment pressure is accepted, and part of the current licensing basis for existing plants.

If one was to ask the question, though, what change in risk has occurred as a result of some new need to call upon the operators to maintain containment pressure, how is that - and I know Marty is going to talk about risk - how is that factored into the regulatory position you're talking about?

MR. LOBEL: If it were an operator action, it would be included in either the EOPs or the abnormal operating procedure.

MEMBER RAY: How do you assess the risk of it going awry? Somehow it fails to achieve the required containment pressure that you're assuming exists.

MR. STUTZKE: Well, in our view, the standard suite of human reliability analysis techniques speak well to this operator action.

MEMBER RAY: Okay. So you'll talk about that later.

MR. STUTZKE: Yes, a little bit more.

MEMBER RAY: Okay. That's fine.

MR. LOBEL: Would it be worthwhile to go through the background in view of the questions and other things? That was pretty straightforward, I think, in the White Paper. I could go through it fast.

VICE CHAIR BONACA: Yes, let's do that.

MR. LOBEL: Okay. Well, we already talked about Reg Guide 1.1 a little. It was issued in November 1970. It had only one position, which was that the ECCS and containment heat removal system should be designed so that adequate available NPSH is provided using only the containment pressure prior to the accident. And it also specified that the temperature of the sump water, the suppression pool should be the maximum expected.

Then there were a series of revisions to Reg Guide 1.82. Reg Guide 1.82, Revision Zero, was issued in June of `74, and was concerned with the design of sumps for ECCS. The Staff was licensing plants at that time, and was concerned with the design of the sumps and different utilities were doing actual model tests of sumps, testing for air entrainment and NPSH.

Reg Guide 1.82, Revision Zero, also specified that the assumed blockage of the sump screens should be 50 percent of the total sump screen area. More reviews led to a revision to that Reg Guide 1.82 that was issued in 1985 that incorporated positions that reflected the results of a USI unresolved safety issue, A-43, which was containment emergency sump performance. And it went into more detail, had a lot more positions about the design of the sump, and insulation, and that kind of thing, but it also maintained Reg Guide 1.1 as the cited guidance for containment accident pressure.

The positions in Reg Guide 1.82, Revision One, weren't backfit. The accompanying Generic Letter stated that the guidance developed by the USI should be used for future plant changes.

Reg Guide 1.82, Revision Two was issued in May 1996, and it was the result of work that led up to NRC Bulletin 96-03 that dealt with BWR ECCS strainer blockage. As a result of installing larger ECCS suction strainer screens, and a new debris source term, some BWRs with Mark-1 containments requested use of containment accident pressure. And those requests were approved after a detailed review.

I didn't mention, I should say at the beginning that some BWRs were licensed, initially licensed with credit for containment accident pressure. That's what led to the issuance of the first Reg Guide 1.1.

MEMBER BANERJEE: But they were grandfathered. Right? I mean, they still have that as license -

MR. LOBEL: It's just part of their licensing basis. And when they put in the larger strainers, some of them, and with the new source term, some of them needed even more pressure than what they had had before.

In 1996 and 1997 there were a series of LERs and other operational experiences that led to some questions about the use of overpressure for operating plants, and the NRC issued a Generic Letter 97-04 in October of 1997, where we asked the industry, asked licensees for information about how they were using containment -- whether they were using containment overpressure and how they were using it.

The Staff reviewed all those responses. When some licensees went back and looked at this issue, they found that their analyses weren't correct, that they made incorrect assumptions in some places, they found problems when they went back and looked at their analysis in terms of assumptions they made for screen flow resistance and that kind of thing. So some licensees with that Generic Letter had to redo their accident analysis for NPSH.

Reg Guide 1.82 Revision Three was issued in November of 2003, and it incorporated guidance that came from NRC Bulletin 2003-01, that dealt with the PWR sump screen blockage issue, and GSI 191. It was at that time that we included NPSH guidance in Reg Guide 1.82. And the idea was that we had come up with criteria that were used for review of the Generic Letter, but weren't published, separately accepted individual plant SERs, so we put the NPSH guidance into the Reg Guide.

And, finally, we started work on a draft Reg Guide 1.82, Revision Four, and we discussed this with the ACRS. And the ACRS wrote a letter to the Staff, either the Staff or the Chairman, in 2005 that recommended revisions and further restrictions on the use of containment accident pressure prior to issuing. And like I explained, we haven't tried to rewrite the Reg Guide.

The NRC position on this issue is that the NRC allows use of containment accident pressure in determining available NPSH when analysis using conservative assumptions have demonstrated that the pressure will be available for postulated design-basis accidents, and when examined from a broader perspective beyond design-basis accidents, an acceptable level of safety is maintained.

MEMBER CORRADINI: Can I ask just about

that historical point? So those conservative assumptions are not documented; that is, on a case-by-case basis the containment analysis that minimizes that pressure has to be kind of discussed with the Staff. There is not a set laundry list of assumptions. Is that correct?

MR. LOBEL: That's true, basically. But, actually, for the BWRs the analyses are pretty much the same, same set of assumptions are used for each one. There are some little changes from one to another that get reviewed.

MEMBER CORRADINI: And the conservatism, of course, is to drive the pressure as low as possible for this analysis.

MR. LOBEL: Well, two things; to drive the pressure as low as possible - well, not as low as possible - to drive in a conservative direction to the point where everybody agrees it's conservatively lower. But the important thing is for BWR, and for PWR, is the temperature of the water. And the analysis at the same time tries to minimize the pressure and maximize the temperature of the water. The temperature of the water is important mostly because if you look at the equation for NPSH, it has a minus vapor pressure term in it, and that's a non-linear term. And as the temperature gets higher, vapor pressure gets much higher.

MEMBER CORRADINI: Okay.

MR. LOBEL: And that's the limiting point.

MEMBER CORRADINI: But that still all goes to containment analysis, because the only way to affect that is besides an uncertainty in decay heat, is what the efficiency of the heat exchanger is that's pulling the heat out of the system.

MR. LOBEL: The efficiency of the heat exchanger is a very important input.

MEMBER CORRADINI: So are those -- back to my original question. Are those assumptions pretty much the same in all BWR analysis relative to heat exchanger, heat losses to cold surfaces, a decay heat so that -

MR. LOBEL: The values aren't the same, but the assumptions are basically the same. Every plant has heat exchangers with slightly different levels of fouling, and tube plugging, and that kind of thing.

MR. DENNIG: Rich, isn't it true that the core analysis has been done usually by the NSSS vendor in most cases?

MR. LOBEL: Well, for this case it's the

containment. There's a core analysis that's done, a mass and energy release analysis, but it's done by the vendor usually.

MEMBER BANERJEE: You mention in your White Paper that containment pressure is needed to -- in some cases, should not exceed Appendix K limits. Can you tell me a little bit about that, because the first peak is usually when the floor is choked, I mean the system is choked, so why does the -

MR. LOBEL: It's during the re-flood that you take credit for containment pressure.

MEMBER BANERJEE: But not the blow-down.

MR. LOBEL: Not the blow-down, no. It's during the re-flood, because the higher the containment pressure, the faster the core re-floods, so Appendix K requires that the containment pressure be conservatively minimized.

MEMBER CORRADINI: But back to my original question. Those assumptions for Appendix K are not the same assumptions used to do this.

MR. LOBEL: No.

MEMBER CORRADINI: Okay.

MR. LOBEL: Containment analysis isn't covered by -

MEMBER CORRADINI: Okay.

MEMBER BANERJEE: They are minimizing the pressure.

MR. LOBEL: They're actually pretty much the same assumptions.

MEMBER CORRADINI: They're both minimizing it, but how they're forcing them to minimize it is not the same set of assumptions. That's my point. They're both minimizing, because if you minimize, you have more bypass for the Appendix K calculation.

MR. LOBEL: A lot of them are the same. If you look at the -- there's a Standard Review Plan section on minimizing the pressure for LOCA, and a lot of assumptions that the guidelines in the SRP are the same as what's used for the NPSH analysis.

MEMBER CORRADINI: All right. Thank you.

MEMBER RAY: I want to just add to what Mike said. I think the issue will come down, there's an awful lot of rhetoric here, not misplaced, but a lot of it that you have to wade through dealing with the issue, is it okay to use containment pressure, when the real issue on the table, I think, is the demonstration of conservative assumptions. So trying to get to that place where we say yes, we have demonstrated conservative assumptions were used, is what we're all struggling with here. And we're having to wade through an awful lot to get to it.

MR. LOBEL: Well, there's tables in the White Paper, a list of assumptions, conservative assumptions.

MEMBER RAY: Well, we can argue about whether they're conservative or not, for example, but that's not my point now. I'm just saying that's the focus of interest, I think, on many of our parts, is a demonstration of conservative assumption, as opposed to the strawman, which is, can we use containment pressure?

MR. LOBEL: Well, the conservative assumptions are used for the LOCA analysis. That's a design-basis analysis. Conservative assumptions are not used for these other events, the Appendix R, ATWS, and Station Blackout, because they're not design-basis events, and the Staff guidance has always been to use realistic analysis. All the ATWS, Appendix R, and Station Blackout analysis is done realistically, not just -

MEMBER RAY: As you wish, but non-conservatively.

MR. LOBEL: Non-conservatively.

MEMBER BANERJEE: So when it's needed for LOCA to meet your Appendix K requirements, how long, and how high?

MR. LOBEL: Not for very long, just until re-flood, until the core is quenched, just until the re-flood.

MEMBER BANERJEE: Relatively short time.

MR. LOBEL: Relatively short time, right.

MEMBER BANERJEE: And how much?

MR. LOBEL: Oh, I can't give you value off the top of my head, but probably around the same value as -- well, maybe a little higher than what's used here. Actually, some licensees, if they can, make the conservative assumption that they don't need containment accident pressure, and they assume that pressure in the containment doesn't change. And they're able to demonstrate that they can quench the core without that. But, again, the -

MEMBER BANERJEE: This is a PWR problem, mainly. Right?

MR. LOBEL: Yes, it's mostly PWR, although not entirely. Okay?

Okay. As ACRS has pointed out, the Staff position doesn't explicitly mention the duration of using containment accident pressure or the amount of pressure used. The duration of the use of containment accident pressure we feel is not risk significant, since the significant contributors to a loss of containment integrity occur at the start of the postulated accident, either pre-existing leak or failure of containment isolation.

MEMBER STETKAR: Let me ask you about that, and Marty may pick it up. Are you going to pick it up more in the risk -

MR. STUTZKE: I hadn't planned to, so ask now.

MEMBER STETKAR: Okay. Let me ask this. Because in the risk part of the White Paper, that sounds like an innocuous assumption, but it's really important because there are conditions where there is no pre-existing leak, and there is no failure of containment isolation, and yet you may have a high likelihood of not having adequate net positive suction head if, for example, the operators don't do something.

MR. STUTZKE: Right.

MEMBER STETKAR: And a lot of the things in the risk assessment are based on this premise that these are the only ways that you can have inadequate net positive suction head. So I was -- I had to wade through a lot of things to figure out how there's an equation in there. I had to wade through a lot of things to figure out how that equation, some of the risk insights work. And it only works if this is absolutely true, if they are the only ways.

MR. STUTZKE: Well, I would disagree with that. It's fair to say that what we've examined so far, these were the only ways. But I believe back in -- I talked about the need to reconsider the human reliability analysis, and so you would have to add in, for example, an operator failure to turn drywell coolers.

MEMBER STETKAR: Right. And that's, for example. That may have a lot higher likelihood than either of these things.

MR. STUTZKE: No, I don't think it does.

MEMBER STETKAR: Okay. That's an opinion, but it might.

MEMBER RAY: That's right. It's a matter could be demonstrated, is all. It's not -

MR. STUTZKE: That's right.

MEMBER STETKAR: And Harold said, it's - what is the conservative analysis? Okay. Continue.

VICE CHAIR BONACA: The White Paper is focused very much on initial conditions, in fact, assuring that programs assure that you have containment isolation at the beginning. There isn't much of a discussion regarding tests being done to address potential degradation of seals, for example.

MR. LOBEL: Well, I was going to talk a little bit about that now, too.

VICE CHAIR BONACA: Okay.

MR. LOBEL: I think we're going to -

MEMBER ABDEL-KHALIK: But that goes directly to the statement that you're making. A failure of a polymeric seal depends on the time at temperature. Wouldn't this statement be incorrect?

MR. LOBEL: If it did, yes, it could be incorrect, if that were another mechanism for failure.

MEMBER ABDEL-KHALIK: Do we have data that show that that is not true, that would support this sort of totally blanket statement?

MR. LOBEL: Well, I was going to talk about that later, and I think Dr. Powers already has a comment on that. I think that's going to be a good part of the discussion here in just a couple of slides.

MEMBER ABDEL-KHALIK: All right.

MR. LOBEL: The possible exception that I identified was in the case of an Appendix R fire, the associated circuits could be a problem at any time during the event, but that is looked at as part of any license amendment that deals with Appendix R. The Staff would review that as part of an Appendix R review, and also independent of containment accident pressure, it's also called multiple spurious operation issue as part of the fire protection closure plan. But the basic response, I think, would be that that issue of associated circuits would be looked at on a plant-specific basis for any license amendment request that dealt with fire.

The magnitude of the pressure it needed is also not included in the Staff's position, since we feel it's not risk-significant. There is a calculation of peak LOCA containment pressure that demonstrates that the pressure is below the design-pressure of the containment, and the pressure at the time of peak sump or suppression pool temperature is much less than the containment design pressure.

VICE CHAIR BONACA: So the issue here has to do with the amount of credit for back-pressure?

MR. LOBEL: Well, I guess -

VICE CHAIR BONACA: It is. So what you're saying is that the amount of credit for back-pressure is not risk-significant. And the question I have is, how can you make a flat statement? Doesn't it depend on how much credit is needed?

MR. LOBEL: Well, I guess the point I'm trying to make, and maybe I'm missing something, is that if the pressure is much less than the containment design pressure, it's hard to see why pressure would be significant. The pressure is there. The operator, in general, doesn't do anything to increase the pressure, or decrease the pressure. Well, I won't say decrease. He doesn't do anything to increase the pressure. He doesn't add mass to the containment. He follows his procedures for cooling down after an accident, but the pressure is there. And it's much less than the peak pressure, the design pressure or the peak pressure, so it's hard to see, unless I'm missing something, why the magnitude of the pressure is important.

MEMBER ABDEL-KHALIK: I guess, if I were to write the first bullet to capture what you're trying to say, I would say as long as the pressure is below the design pressure, the magnitude of the pressure needed is not risk-significant. Is that what you're trying to say?

MR. LOBEL: That's what we're trying to say.

MEMBER ABDEL-KHALIK: Okay. Now, the implication, of course, if I have two accident scenarios, and the pressure history in both accident scenarios, the peak pressure is the same, but one of them, the duration of the high pressure in both cases, the peak pressure is below the design pressure, that regardless of how long this peak pressure period is going to be, the risk from these two scenarios is going to be the same.

MR. LOBEL: Well, this slide is talking about the magnitude. The previous slide was talking about the time. And that gets back to the discussion of the duration of the event.

MR. DENNIG: I think there's two, the pressure goes two ways here. I think the pressure that you're trying to talk about is that there's no threat to the containment, where you're going to lose the containment, because the pressure is too high at that point.

MR. LOBEL: Well, there's no threat on -

MR. DENNIG: And there are conservatisms in the calculation for the pressure for the NPSH calculation that go into that margin.

MR. STUTZKE: Let me jump in for a minute.

MR. LOBEL: Yes.

MR. STUTZKE: Because if you look at the types of failure mechanisms that could lead to a loss of pressure in the system, for example, containment isolation doesn't isolate. You either have the pressure or you have no pressure at all. It's hard for the PRA to distinguish between a scenario that says I need 3 psi overpressure for 10 hours, versus 3-1/2 psi for 12 hours. I couldn't compute the delta risk in that and convince you guys that the number was right. Okay?

CHAIR SHACK: But in terms of defense-in-depth for my unknown, or say my sump blockage issue, I am reducing my defense-in-depth here. Again, something that I'm really fairly highly uncertain about, which are my head losses and sumps during accidents.

MEMBER BANERJEE: Which is, of course, a point we should visit eventually, the uncertainties part of this.

MEMBER APOSTOLAKIS: But it seems to me that this slide and the previous slide are the basis for the risk assessment, where you're really evaluating only the existence or non-existence of containment overpressure.

MR. STUTZKE: Yes.

MEMBER APOSTOLAKIS: That's all you do there.

MR. STUTZKE: Yes.

MEMBER APOSTOLAKIS: When we say risk assessment, that's all you do. And the basis for that is these two slides.

MR. STUTZKE: That's true.

MEMBER APOSTOLAKIS: Okay.

MR. STUTZKE: That's true.

MEMBER CORRADINI: So maybe I'm getting ahead of you, so you can tell me to stop. But I'm kind of -- what I hear in all the discussion is saying that - what you said, Marty, at the end was, is that I can't tell the difference between 3 and 5, and 10 hours and 20 hours. As long as it's high enough that I don't kill the pumps by bad performance, and low enough that I don't fail containment, or exceed design pressure.

MR. STUTZKE: Right.

MEMBER CORRADINI: But that almost says now I've crossed the two defenses-in-depth, and I'm going to sit here with a magical regulator that makes it just high enough to make it work, and just low enough not to fail containment. That's what I hear in terms of the behavior. Am I mishearing?

MEMBER APOSTOLAKIS: So what?

MEMBER CORRADINI: Well, that tells me I

have a very intelligent operator that can sit there with a dial to dial it in between the two.

MR. LOBEL: The operator doesn't have to do anything but follow his procedures, and his procedures are basically to cool the plant down, but make sure that he -- cool the plant down, cool the core, cool the containment, but make sure he has adequate NPSH. And that's only -- he doesn't really have to take any actions. All he's doing is looking to make sure he's okay.

MEMBER CORRADINI: Okay. But let me push back one more time. So let's just take a few numbers. So let's say I needed an NPSH of -- I needed a delta above P-zero at time-zero of 10 psi, and the design pressure is 40. That means I've got a 30 psi D dead band that I have to operate in, and the uncertainty of whether it's 10 or 15, or it's 40, or 38, and as I approach it, and that uncertainty is nowhere in this, so that it would seem to me that the procedures could get fairly complex.

MR. LOBEL: The operator -- whatever is happening in containment is happening in containment pretty much, and if he has 40 when he needs 10, all he's going to look at is I have 40, and he might -- and in a BWR, he might look over at these curves just to make sure he's okay, but that's all he'd do, and he'd satisfy himself that he was okay. He doesn't have to take any actions.

MEMBER SIEBER: If he's supposed to get 40 but only gets 30, there isn't anything he can do about that.

MR. LOBEL: That's true, too. Well, there's -- not necessarily. Not necessarily. And we'll just talk about that a little later, too.

MEMBER RAY: I'll defer to you here in a second, Otto, but I'm going to piggyback on something Sanjoy said. The first observation is, you're saying well, this is just more of the same. The operator does all these things. He can do this next thing as part of the EPU package. Okay. I understand that argument.

It still raises the question of what's the incremental risk involved, because you can't tell me that there isn't some risk of operator error. If that's what you're trying to say, then we can stop there.

The other thing, though, is the uncertainty associated with something, let's say the heat transfer to the plant structure. It seems axiomatic that we're trying to minimize the assumed pressure loss due to heat transfer when we're concerned with the containment integrity question.

This is the opposite. True? In other words, true enough, as Jack says, he may have insufficient pressure, and there's nothing he can do about it, because we've missestimated what the loss of pressure would be due to heat transfer, and this gets to Said's question about time. Time is relevant here.

MEMBER BANERJEE: If you've missed Jack's last point, you can head for the parking lot.

MEMBER RAY: Well, I'll be done in a second, because I realize I'm lecturing rather than asking a question. But my point is, how is it that we do all the right things in assuming not too much temperature heat transfer, and therefore maximize the pressure from a containment integrity standpoint, and do the opposite when it comes to heat transfer that may result in a loss of pressure.

MR. LOBEL: Well, in real life, if you had one of these events, the operator would follow his procedures, and the plant would do what the plant does. The operator isn't controlling that kind of thing. The plant is doing what the plant does.

MEMBER RAY: You have a discussion in there about, you referred to it a minute ago, about looking at this pretty variable graph and throttling, and operating the system. Okay. Granted, all of that's there. But then, ultimately, I'm going to ask you, all right, what is the incremental risk then that's associated with this extra task that has now become required for the operator to perform?

MR. DENNIG: Rich, don't we do a different calculation to maximize the heat content of the water to drive the temperature up for the NPSH calculation, and then we do -

MR. LOBEL: Well, we're trying to separate -

MR. DENNIG: -- the other way around to maximize the pressure.

MR. LOBEL: I'm trying to separate -- there's two things, there's what would really go on. There's what would really go on in the real world, and then there's the way the analysis is done. The analysis is done in a way that actually minimizes the heat transfer around the suppression pool. For example, you don't take any credit for heat transfer from the torus wall to the reactor building. That's assumed to be adiabatic because what that does is it drives up the temperature of the suppression pool, which is conservative in the calculation. In the real world, there's going to be heat transfer to the reactor building, and the operator has no control over that.

MEMBER RAY: Yes, that's so, but on the other hand, we're looking at pressure margins here which are really tiny. Even assuming the operator is perfect all the time, it's a tiny margin.

MR. RULAND: What are you looking at?

MEMBER RAY: The question is how much uncertainty should be assumed?

MR. RULAND: What are you looking at?

MEMBER RAY: I'm just looking at a table of pressures versus different operating conditions.

MEMBER BANERJEE: I'm having a much more fundamental problem, and maybe Marty can address this. From what you're saying, is as long as I let my containment pressure stay below the design pressure, it doesn't matter for how long. There's no incentive for me to actually try to reduce this pressure. Everything gets easier if the pressure is high; obviously, the steam is more compressed so my RHR system pumps less steam out, the pressure losses are less, so why do I try to reduce the pressure? By your argument, I'll simply keep it as high as I can below design pressure.

MR. LOBEL: The problem is I think here we're focusing on this one issue. The operator in the control room after the LOCA has a lot of other things to worry about, and he is trying to shut down -- he is trying to cool down the suppression pool, and the core.

MEMBER BANERJEE: By the risk analysis it doesn't show up. Right?

MEMBER SIEBER: Well, the idea is that the operators will bring the plant to a cold shutdown.

MR. LOBEL: Right. And he's going to do that. And the only place that this comes in is, he's going to check, he has a caution in his procedures to make sure that he doesn't cool down more than he should, but that's just one -

MEMBER RAY: Wait a minute. No, no, that's not fair. He's got to secure these doggoned drywell coolers in, what is it, two hours I guess it is. And if he doesn't do that, the long-term pressure that Said is worried about isn't going to be achieved.

MR. LOBEL: But he has other operator -- the operator has other actions he has to take, too.

MEMBER RAY: That's why I said, you're just saying well he does more of the same thing.

MR. LOBEL: Well, at some point, this is

not a reasonable thing to assume.

MEMBER STETKAR: I think what Harold is getting at, and what you said, the operator does have a lot of other things to do. However, all of those other things are pointing at a certain trajectory; that is, cooling the core and reducing the heat load inside the containment. That's what all of those other actions are pointing him to do. This is contrary to all of those other actions. This is contrary to all of those other actions.

MR. LOBEL: He has -

MEMBER STETKAR: There's a specific focus only on the containment. So, yes, he does have other things to do, but at least they're consistent.

MR. LOBEL: He already has an action -

MEMBER SIEBER: You have to cut down the rate of cooling. That defeats the purpose.

MEMBER STETKAR: That's right.

MR. LOBEL: The operator has an action -

MEMBER SIEBER: That's a different kind of activity.

MR. LOBEL: The operator has a similar action to turn off the containment sprays before he gets to a certain point.

MEMBER SIEBER: That's -

MR. LOBEL: I'm sorry?

MEMBER SIEBER: If you don't cool the core, once you start cooling water inside -

MR. LOBEL: Let me just -

MEMBER SIEBER: -- in circulation mode, once you -- that's your only way of removing heat from it.

MR. RULAND: This sounds like what we've heard before as the counterintuitive argument, that somehow operators -- that this whole notion of cooling the plant down and stopping at some juncture is somehow counterintuitive. Well, in 1979 there was a small accident that occurred at Three Mile Island where the operators, in fact, acted on their intuition. They had an intuition that the plant was going to go solid. And, in fact, the plant wasn't going to go solid. Okay? They were acting on their intuition.

Since Three Mile Island, what has happened is, we have trained operators to obey their procedures. And, in fact, we use simulators that they didn't have, so this notion that operators somehow can't operate a nuclear plant in accordance with their procedures, I don't believe has the merit that I think I'm hearing from the Committee.

MEMBER RAY: Well, I just -- I don't think - I don't agree with you at all.

MEMBER STETKAR: Let me step in here, because I was -- I happened to have a Senior Reactor Operator's license and was operating a pressurized water reactor at the Zion Nuclear Power Plant in March of 1979, and had been through license training, and had procedures. And one thing that we were very well trained on was do not let the pressurizer go water solid. So this idea that the operators were acting on their intuition is not quite accurate, because they had high level indication in the pressurizer. They had all the indications available to them that they had adequate inventory, and adequate core cooling, and they acted exactly in response to their training to put the plant in what they thought was a safe condition. So it's not just guys acting on their intuition. They were acting according to their training.

(Simultaneous speech.)

MEMBER STETKAR: -- to putting the plant on a trajectory towards what they thought was safe.

MEMBER BROWN: But we also teach operators, at least we did, to believe your instrumentation.

MEMBER STETKAR: That's right. That's why I'm saying, your instrumentation -

MEMBER BROWN: I'm just throwing that in as part of the -

MEMBER RAY: But this is a question not of believing instrumentation, but of relying upon the operator to intervene.

CHAIR SHACK: Maybe we can move on, because Marty is going to have to come back to address this as an operator error, as part of the risk.

MEMBER MAYNARD: I've been patient, and this is not about operator action. This is about the first statement up there on the risk significant. We seem to focus primarily on risk to the containment, and what we're really interested in is risk to the public. And one of the areas is that if you do have a leak in containment, the longer you stay at a higher pressure, the more risk-significant it is.

(Simultaneous speech.)

MEMBER SIEBER: When we finally come back to it, I want to complain a little bit about time duration not being reported. It depends on the -

MR. LOBEL: That's true.

(Simultaneous speech.)

MEMBER APOSTOLAKIS: That was really my

question. I don't understand -

(Simultaneous speech.)

MEMBER APOSTOLAKIS: I mean, I didn't see time anywhere. Time is brought in as convenient. Here are some tests. The ACRS recommended a couple of hours. On the next slide, for example, where you have the -

CHAIR SHACK: Jump to the next slide. This is your cue.

MEMBER APOSTOLAKIS: So if I don't know anything else, I look at this, so if the NPSHA is less than the NPSHR -

MEMBER SIEBER: I'm not -

MEMBER APOSTOLAKIS: But if it becomes an inequality, then presumably I'm in trouble.

MR. LOBEL: If it's less -- if A is less -

MEMBER APOSTOLAKIS: And a lot of people come in and tell me oh, but it happens only for two hours, you're not really in trouble.

MEMBER SIEBER: It depends on how much and how long.

MEMBER APOSTOLAKIS: That's what I'm trying to understand. What is the failure criteria?

MEMBER SIEBER: It depends on the pump.

MR. LOBEL: The problem with time and pressure is that they're not really addressing a physical process except maybe for the pump time duration.

MEMBER SIEBER: Yes.

MR. LOBEL: The physical boundary is more the required NPSH. That's what the pump vendor tells the pump user, is a place where you can operate, or you really should operate above that. But you can operate there, and you will get a reduction from 100 percent flow, and 100 percent pressure. If you realize that, you can operate there. And it gets into the type of pump.

MEMBER BANERJEE: And it gets into the flow rate, your pump curve.

MR. LOBEL: But that determines the required NPSH.

MEMBER MAYNARD: One thing that would help me in this, I don't think -- I think that it would be better if we went and redefined the NPSH required. And I don't care if you do it on a pump-by-pump basis, or plant-by-plant, whether it's one pressure or temperature for this duration, and it changes. I don't think we should have a position where NPSH available can be less than NPSH required. And I think the way to do that is to not just use what the pump vendor comes out with NPSH required, is if you have solid justification for operating at a different level, let's change that to be the NPSH required.

MEMBER SIEBER: That's hard to do.

MEMBER MAYNARD: Now you're ending up with okay, it's okay to violate it for a while, but there's no -

MR. LOBEL: In a way that's what's done. If you look back at the curves that the pump vendors supplied to Vermont Yankee and to Brown's Ferry, that's what they did, really. They went below the 3 percent head drop, which is the Hydraulic Institute standard, and said you can operate below that for this length of time, and then you have to increase it. You can operate a little higher for that length of time. They did that kind of thing, and they did just what you're saying. They redefined the required NPSH.

The other thing that's done that we've allowed is, we've allowed people -- their analysis show that they're in cavitation for a certain amount of time, we said that was okay, if you go out and test your pump and show that your pump will operate for that length of time with no adverse consequences. And what people have done, and there's a table of people who've done that kind of testing, what people have done is they go out and they do a cavitation test for much longer than the time they're crediting, and then they take the pump apart, and they actually look at the seals, and shaft, and bearings, and everything, impeller, and show that there wasn't any damage, and no wear for even a longer time than what they're crediting. And Staff has allowed that, also.

MEMBER BROWN: But that's a new pump.

MR. LOBEL: Well, it's a new pump. These pumps are tested quarterly -

(Simultaneous speech.)

MR. LOBEL: They're not cavitation tested any more.

VICE CHAIR BONACA: What troubles me with this is that all the discussion essentially is attempting to make cavitation an acceptable mode of operation. And, clearly, that was never intended in the design of these pumps. It was never intended when the original analyses were made, so I want to say that when I hear this conversation and I look at all the arguments, pumps are tough, they can take it, whatever. Why should we allow for that to happen, there is this degradation in performance, and it's considered to be acceptable. I mean, that's what really what I struggle with.

MEMBER APOSTOLAKIS: For voluntary -

VICE CHAIR BONACA: For voluntary applications. I want to get power, I spend a little money to uprate my plant, but I cannot invest anything on these pumps.

MR. LOBEL: Well, the pumps don't operate at this condition for the whole accident. They don't operate at this condition when they're not in this accident. This is just when they're calculating that -- this is a criterion for calculating that they need containment pressure for NPSH.

VICE CHAIR BONACA: I understand.

MEMBER APOSTOLAKIS: Is it true that this is a sufficient condition, but not necessary? That's really what you're arguing. It's sufficient to avoid cavitation, but it's not necessary. In other words, you can violate it for a while.

MR. LOBEL: It's more complicated than that. When you're at this point, you are cavitating a certain amount. You're cavitating, and you have a 3 percent drop in head. You're cavitating when you're above the 3 percent drop in head, when you're at 1 percent you're cavitating. Way down the pipe somewhere you could be cavitating, even when you're not noticing that you're cavitating, you can have some. And there are pump studies that show that even a little bit above this criterion is actually worse. You can actually get more damage to the pump than at this criterion where the available is equal to the required.

MEMBER APOSTOLAKIS: That's true.

MR. LOBEL: People have done testing, pretty reliable testing, where they've used noise measurements to show that the -

MEMBER SIEBER: Sometimes you're better off cavitating -

MR. LOBEL: Yes. You're actually better off cavitating the 3 percent, than to go to some smaller margin above the 3 percent, because you could actually be doing more damage to the pump.

(Simultaneous speech.)

MR. LOBEL: And I can give you references that talk about that.

CHAIR SHACK: In George's terms, though, this is an acceptance criterion for this analysis.

MR. LOBEL: It is an acceptance criterion.

CHAIR SHACK: So I think we can move on. Now, you've made an argument why it's an acceptable one, but can we -

MR. LOBEL: Yes. Let me -

MEMBER BANERJEE: By that argument, we could suck a little air in, but you never get cavitation.

CHAIR SHACK: As long as he's got this, he can do this forever.

MEMBER SIEBER: Why don't we just move on.

CHAIR SHACK: Okay. We'll come back to when this is violated, but this is the top level requirement that they have, and then they make some exceptions.

MR. LOBEL: We're running out of time.

(Off the record comments.)

MR. LOBEL: A lot of what I was going to say I've said before.

VICE CHAIR BONACA: Special initial conditions is creditable. I mean, I believe -

MEMBER APOSTOLAKIS: But it's also true that everything that was said in the last few minutes did not address your concern. Your concern is different. Why should we accept even a little bit of cavitation just to accommodate a voluntary change in power?

MEMBER SIEBER: Don't we have instances come up in the near future where there is no change in power involved, the credit for containment pressure is going to be required due to 191.

MEMBER RAY: So that raises a question, should we use the same margin more than once. Before you go on to the next slide, though, I just want to again say I'm not comfortable with where we are on this LOCA pressure is conservatively calculated question, particularly when you talk about it over a long period of time. I just want to register that, and then move on.

MR. LOBEL: Could you say -- I thought -

MEMBER RAY: Some up there that says this is acceptable because LOCA pressure is conservatively calculated.

MR. LOBEL: And there's a whole list of conservative assumptions that are used in these -- I don't understand, I guess -

MEMBER RAY: Why I say that?

MR. LOBEL: Yes. What's conserving -- I mean, I'd like to answer your question, and I'm not sure -

MEMBER RAY: Yes, but the Chairman wants us to move on, and I don't disagree at the time that we need to move on. You can either through inadvertence, or just through misestimating end up with more heat loss than you assume, unless you do something to prove that this is a conservative calculation, and I just haven't seen that. The list you've got in your White Paper doesn't do it for me. That's it.

MR. LOBEL: Well, you overestimate the power, you overestimate the decay heat, you overestimate -- you underestimate the capability of the heat exchanger, you don't take any credit for heat transfer from the torus to the outside.

MEMBER STETKAR: That keeps pressure high. Doesn't it?

MR. LOBEL: Well, it keeps the pressure high, but the big effect is the temperature. GE -

MEMBER RAY: That's fair, but we've got to see it, not just have it be on the list is my point.

MEMBER BANERJEE: So the pressure is not -- it's keeping it high, but the water temperature is higher, so you've got -

MEMBER RAY: The argument is that they're more than compensated. I just want to see numbers, Sanjoy.

MEMBER BANERJEE: I agree. I mean, we need to see a proper answer in the analysis of all the -

MR. LOBEL: GE has done those types of calculations where they've looked at the point of minimum pressure, and the point of maximum temperature, and they've shown that the available NPSH is much lower at the point of maximum temperature.

MEMBER RAY: Well, you have a lot of information that we don't get, evidently.

MR. LOBEL: Okay. Well, confidence in containment. These are the points that I was going to make for the technical basis. Let me just say the containment is tested prior to operation, Appendix J leak testing is done, 50.55(a) in-service inspections are done, tech specs require various things to assure containment integrity. Plants do checks in their procedures prior to start-up to assure they have containment integrity. Conservative calculations, I guess we've talked about that.

Pump design, I think we covered a lot of that under margin. I think we've already talked basically about that. I think we've talked about emergency operating procedures, and plant risk is Marty. I'll turn it over to Marty.

MEMBER ARMIJO: Before you go, on pump design, Chart 20 that you have, can you put that up. Yes. In reading your paper, which, incidentally, I thought was very well written. I learned a lot about this stuff that I didn't know about. There was a table that showed that there were, in fact, more cavitation resistant impellers available, aluminum bronze had under the same conditions twice the cavitation resistance of stainless steel. And why isn't that considered to improve the margin for the pumps?

MR. LOBEL: You have other considerations, too.

(Simultaneous speech.)

MEMBER ARMIJO: -- you get in trouble. That's in a severe -

MR. LOBEL: In a PWR you have to worry about boric acid.

MEMBER ARMIJO: How about just the B?

MR. LOBEL: And in a BWR -- well, the -- what was I going to say? The difference isn't that great, and you don't expect the pump to operate in cavitation for a long time. We're talking about matters of hours, we're not talking about material that you want to pick for a chemical processing plant, or a power plant, and other applications where that pump may be running continuously for six months or a year, or more.

MEMBER ARMIJO: Yes. I just -- it was a very simple table. Said hey, look, here's a number of hours which stainless steel operates and is damaged, and you can go twice as long with this aluminum bronze. And it seems like why wouldn't that be a good thing to do, if you were concerned about cavitation damage, whether it's short time, or long time.

MEMBER SIEBER: Generally, what you -- you're not so much concerned about wear to the impeller, break the impeller, or ruining the pump bearings. Vertical shaft pumps, long shafts. When they cavitate, they beat the hell out of the bearings, whereas, the horizontal pump with a short shaft won't do that. If you run into cavitation for six hours or eight hours, something like that, you're not going to wear through the impeller, but if the impeller isn't strong enough, you may break it.

MEMBER ARMIJO: So these are the best pumps available. We can't do anything better.

(Simultaneous speech.)

CHAIR SHACK: In fairness, if they're going to operate in the cavitating mode, they ask for tests to demonstrate that they can do it. So you may be able to get a better pump, but they ask for a demonstration that it can be done.

MR. LOBEL: Okay. We haven't talked about seals. Can we spend a minute talking about seals?

CHAIR SHACK: Seals I think are a major interest item.

MR. LOBEL: That question came up a couple of times. I looked at the paper that Dr. Powers referenced before. I looked up another paper, looked up some other papers, and I talked to the people who do this review. And my understanding is well, first of all, that this is something that's looked at as part of any review, like an EPU or something, where there's a question of environmental qualification, mechanical equipment qualification is part of that review. That's more a programmatic review to see that these things have been addressed, that the licensee has actually thought about elastomer seals, air lock door seals, or whatever, and that the licensee states that the seal is qualified for whatever harsh conditions the licensee calculates that he's going to have.

From what I could see, the seals start to not do their job for severe accident-type conditions, that they seem to be okay for design-basis accident conditions. And if that's not correct, I guess -

MR. RULAND: Rich, it's not seem to be.

Right?

MR. LOBEL: That seems to be the case.

MR. RULAND: Licensees are required to have an environmental qualification program, and required to demonstrate for the LOCA envelope that they have assumed that the seals will withstand the environmental conditions, so it's not a question of seems.

MR. LOBEL: They make that statement, and the Staff reviews that as part of EPU reviews, or other LOCA analysis reviews, anything that's going to affect those conditions. Do combinations, you know more about that than I do. I don't know the answer off-hand.

MEMBER POWERS: I know very little about the subject, except what I've imparted to you. What I know is a couple of things. I know the paper I referenced to you from the Japanese, they've observed an interesting phenomenon that the combination of irradiation and steam was by far the harshest environment, and was not replicated by doing either steam alone, or nitrogen plus irradiation alone. It is substantially worse.

The other thing I know is that elastomers are subject to loss of elasticity in a radiation field, and it depends on the elastomer that's used. And the third thing that I know is work sponsored by the NRC on the degradation of organic materials found a synergism between temperature and the radiation field; that is, the existence of heat and radiation was worse than either by itself.

MR. LOBEL: Was that significant in the range of design-basis accidents, or more in the severe -

MEMBER POWERS: I told you I would tell you what I know, but I exhausted my inventory.

(Laughter.)

MR. LOBEL: The Japanese paper, I believe, used a radiation source term for severe accidents, not for design-basis accidents.

MEMBER POWERS: Well, I guess the question I was going to pose to you, what I know is that most of the environmental qualification tests presume about a megarad per hour in the atmosphere, and about two megarads per hour at solid surfaces. Those are fairly formidable doses. There's not much difference, in other words, between a dose from a severe accident and a DBA.

MR. LOBEL: Well, if I remember right, this paper was using 85 megarads for their dose.

MEMBER POWERS: Pretty hard to be at.

MR. LOBEL: Yes.

MEMBER POWERS: I don't know how they would do that.

MR. LOBEL: Yes.

MEMBER POWERS: If you look in the COC Handbook on elastomer's properties, what they look at is the integrated dose. They don't look at the base rate, but the integrated dose. And on a good day, I'd actually be able to quote some numbers to you. I hesitate to do that right now, because so far it's not been a good day. It depends very much on what the elastomer material is.

MR. LOBEL: Well, let just say again, I know I talked to the people who do the reviews. They showed me the reviews they're doing for the advanced plants now, and they do look -- they get a list of the licensee of all the seals, and the conditions that they're qualified for, and the conditions to expect, and they do review that.

MEMBER POWERS: It would be intensely interesting to me to get that self same list, and an example how they qualify the materials.

MR. LOBEL: Okay.

MEMBER POWERS: NRC did presume that cable insulation within plants would not experience an embrittlement degradation, and were stunned to find out that it would, and that's why they sponsored some research on how elastomers and polymeric materials respond to both temperature and dose rate. And they saw significant degradations at integrated dose rates of only 2.5 megarads, which just blew everybody away. And then they found out well, it's because there's a synergism between temperature and radiation. Interesting, and that's all I can say. And I brought it up to you because it struck me as an issue to consider, when you talk about 10-minute times, and 15-minute times, polymer is going to do just fine. When you get up to 91 hours, that's when I start saying wait, I want to know the details on the environmental qualification now.

MR. LOBEL: Well, don't forget for this long time event that we keep talking about, the Appendix R event, there's no core damage for that event.

MEMBER POWERS: Then you have a much lower dose rate.

MEMBER BANERJEE: In this Appendix R event, do you have the potential for entraining sludge and stuff?

MR. LOBEL: No. The only event that had the potential for doing that in what we've looked at together so far was Vermont Yankee for the ATWS event. They relieved pressure with a safety valve that discharged into the containment, and so there was some debris generated with that.

MEMBER BANERJEE: No, not generated. I'm just saying stirring up the -

MR. LOBEL: Oh, turbulence?

MEMBER BANERJEE: Yes.

MR. LOBEL: Oh, as long as the RHR pumps are operating, you have turbulence in the suppression pool. There have been tests that show even with one pump going -

MEMBER BANERJEE: It's sufficient turbulence to -

MR. LOBEL: Sufficient to stir the whole - to not have thermal stratification in the whole suppression pool.

MEMBER BANERJEE: I see. That's interesting. Do they take that into account in the uncertainties?

MR. LOBEL: Where is Zeyna? Hey, Zeyna, we need help.

MEMBER CORRADINI: So can I just -- I was looking through the slides that you skipped over because of time. So the point of all of this is that you still -- the Staff still feels that given that you operate between what you'll call, and I'm still struggling with this, a conservative value for the accident pressure, and the design pressure, as long as that window is large enough, you feel comfortable allowing the operator to throttle back on cooling to maintain a high enough pressure. That's what -

MR. LOBEL: Yes.

MEMBER CORRADINI: Okay. And how small of a window does it have to be before the Staff worries about the uncertainty of allowing that behavior? So in other words, let's forget about NPSH, all that other -

(Simultaneous speech.)

MEMBER CORRADINI: I'm just trying to understand. If I'm within 1 psia, am I getting uncertain about the ability to operate in that window? If I'm within 10 psia? In other words, if I grant you that allowable behavior, at what point does it become too close to call?

MR. LOBEL: The pressure he has, to the pressure where he has to take an action now you're talking about? The operator probably -- the operator is going to operate the plant. He's going to be looking at the flow. He's going to be looking at pump current. He's going to be looking at pressure, discharge pressure if he has it in the control room. He's going to be looking at those things, and if everything is going okay, he's not going to do anything.

CHAIR SHACK: But I think Mike's question is when -

MR. LOBEL: He doesn't have to worry about what the margin is.

CHAIR SHACK: -- you review the analysis, when are you going to say no?

MEMBER CORRADINI: Yes, that's what I'm asking. That's it precisely.

MR. LOBEL: We don't have a -- in the analysis, we don't have -

CHAIR SHACK: You go all the way through the design pressure, as long as it's there.

MEMBER CORRADINI: Yes, all the way to design pressure.

MR. LOBEL: He's not going to be anywhere near the design pressure.

MEMBER CORRADINI: We've seen analysis in certain plants under certain conditions that you get damned close. I mean, excuse my English. Kind of close.

MEMBER SIEBER: Accident pressure.

MR. LOBEL: That's a different analysis. Yes, you're talking -

MEMBER RAY: You're not talking about containment design, are you? He's talking about a pump, NPSHR.

MEMBER CORRADINI: Right. But I saw analysis with certain accidents -

MR. LOBEL: Oh, you're talking about the curve of accident pressure -

MEMBER CORRADINI: Yes.

MR. LOBEL: -- and a curve of pressure -

MEMBER CORRADINI: Yes.

MR. LOBEL: Okay. We don't have a criterion. There's no criterion for how close those can be.

MEMBER CORRADINI: I'll stop.

MEMBER SIEBER: There is one case like that.

MEMBER CORRADINI: Right. And so I'm still troubled by that. That's why I'm struggling.

MEMBER BLEY: And when it seems there might be other sources of uncertainty that haven't been addressed in a conservative way, or even addressed.

CHAIR SHACK: Well, I think it becomes a question of whether you think the conservatisms that he knows he has overwhelm the -- or are sufficient to address the uncertainties that haven't been addressed.

VICE CHAIR BONACA: I think the curve Mike was referring to was the Appendix R.

MEMBER CORRADINI: Yes.

MR. LOBEL: Yes, for the LOCA, I feel confident that there's plenty of conservatism there.

MEMBER CORRADINI: At least I got an answer, so I'm with you, for the point. I'm fine.

MR. LOBEL: Okay.

VICE CHAIR BONACA: All right.

CHAIR SHACK: Before we leave, I want to know what changes you're going to make in the White Paper regarding the risk considerations, so I don't want to run out of time.

MR. LOBEL: That's what we're going to talk about now.

MR. STUTZKE: I will say a practical matter, I've got to get out of here by 5:30. If I don't pick up my kids, my wife will kill me, and there's no uncertainty in that.

(Off the record comments.)

VICE CHAIR BONACA: We'll shoot for 5:15.

MR. STUTZKE: Okay. Our position is that we have adequate guidance in place. We've got processes that implement the guidance, and that we're following it on the use of risk insights in the review of license amendment requests. Okay? And we would propose to continue applying that guidance to those license amendment requests that contain requests for containment overpressure credits. Okay?

I want to spend some time explaining to you what -

CHAIR SHACK: Does this mean that you will amend 1.82 to require them to submit this information as -

MR. STUTZKE: No.

CHAIR SHACK: -- indicated in the White Paper?

MR. STUTZKE: No.

CHAIR SHACK: Okay. What does it mean?

MR. STUTZKE: That's a change. Let me explain what it means.

MEMBER STETKAR: It means if somebody comes in and says I want to use a risk-basis, they will consider it.

MR. STUTZKE: Well, that's one thing. Let's back up and make certain everybody understands. There's no regulation that says an operating plant has to have a PRA. They're used to that in Part 52 space for new plants that will have to have a PRA, but there's nothing now that says they have to have one.

CHAIR SHACK: There's nothing that says we have to give them an EPU either.

MR. STUTZKE: Okay. So license amendments fall into two categories, risk-informed and not risk-informed. Basically, risk-informed license amendments are ones that the licensee declares are risk-informed. He's volunteered for it. He's agreed to follow Reg Guide 1.174, et cetera, et cetera. In most things, the risk insights are one of the principal justifications to say yes or no. Okay?

Those are different than non-risk informed license amendments where a license has not declared that he's following Reg Guide 1.174. Okay. Here the situation gets a little bit different. We refer to Appendix D of SRP Section 19.2. By the way, there's a couple of backup slides there to give you a chronology of how this guidance evolved. It started basically back in 1998. There's a couple of SECY papers and SRMs. There was an ACRS briefing, CRGR Public meeting, and it's been clearly vetted at that time. But the notion is this, is a non-risk-informed license amendment, risk insights are used to decide whether or not you rebut a presumption of adequate protection despite that you're meeting regulations or regulatory requirements.

Okay. The burden is on the staff to demonstrate that we have a question of adequate protection. Okay? Now, we have legal authority to demand the information, as Dr. Schack pointed out, licensing can decline to give us the information, and we can just say can't reach a decision, so you're denied.

MEMBER APOSTOLAKIS: Can you really do that?

MR. STUTZKE: Well -

CHAIR SHACK: Yes, sure.

MEMBER APOSTOLAKIS: What do you mean "sure"? Have you done it?

MR. STUTZKE: We have not done it before, because they always give us the information. It's a couple of slides ahead, but basically to deny a non-risk-informed license amendment on the basis of a risk argument, that's got to go up to the Director of NRR. And in order to that -- but let me -- I'm jumping ahead a little bit like this.

The other thing that's important to realize is that in Reg Guide 1.174, we have five key principles of risk-informed decision making. When you do a risk-informed license amendment request, you're supposed to meet all five of those. If you're doing a non-risk-informed, those principles that are not met are the ones that you drill down on, and you do more analyses. But the mere fact that you don't meet one out of the five, or two out of the five doesn't imply a lack of adequate protection. Okay? Very simple.

I mean, the example is this. The first requirement says you meet regulation. Well, we grant exemptions to regulations, so clearly you don't meet the first condition in that case.

VICE CHAIR BONACA: You must maintain the independence of barrier.

MR. STUTZKE: We have to maintain defense-in-depth.

MEMBER APOSTOLAKIS: Not an exemption, that's part of the regulation.

MR. STUTZKE: Well, there's a regulation that has the process to grant exemptions.

MEMBER APOSTOLAKIS: Anyway, the last sub-bullet, what does it mean; "Assumes that the burden" -

MR. STUTZKE: The burden is on the Staff. And, actually, George, I'm glad you brought it up, because the question came up earlier in our discussion today about how do EPUs fit into this. Well, basically, EPUs fall under Appendix D. In fact, they're culled out as one of those situations that because of synergistic effects, may create special circumstances. In fact, it says EPUs that are significantly above what the Staff has previously approved, if I remember the right words.

Now, a couple of years ago, at the Committee's suggestion, the Staff developed a so-called EPU Review Standard, the guidance on how we're going to go about reviewing EPUs. And if you look in there, you'll find a chapter that says here's the risk evaluation. Okay? And the way that I've looked at it is, the Staff at the time that the standard was developed, had decided that special circumstances may exist, and so we were asking a priori to get risk information, knowing full well that EPUs are not risk-informed, in an effort to expedite the process. Why wait for the Staff to write a bunch of RAIs if the licensees can provide the information in advance? That's the only case that I'm aware of where we have gone to that extent. Everything else would depend on the Staff deciding that we needed to look at the risk-information, and then we would ask questions, and it would escalate that way.

MEMBER APOSTOLAKIS: So, basically, you must show that adequate protection is not there.

MR. STUTZKE: Is not there.

CHAIR SHACK: Or is at least in question.

MEMBER APOSTOLAKIS: The applicant just uses deterministic methods, or Appendix D. Right?

MR. STUTZKE: Right. Okay. Well, here are the times when the guidance of Appendix D is invoked, and it's an interesting language here. It says, "The Staff believes that a non-risk-informed license amendment may significantly change", et cetera, et cetera. Okay? Staff believes. Here's how it works in practice. All right.

A license amendment comes in, it's reviewed by the Project Manager in accordance with our procedures, and it's farmed out to the various reviewers. So somebody like Rick will say, gee whiz, I'm not comfortable with this, something is wrong. Well, we have an internal office instruction that says gee, go ask a Risk Analyst if you feel bad about these things. Okay? All these sorts of criterion, and at that time we would get involved and decide if we wanted to do a review, and the extent of that review, and so forth and so on like this, treat it more or less on a case-by-case basis.

Realize, these situations are extremely rare with the exception of EPUs, and probably the one that kicked it off which had to do with electro sleeving of steam generator tubes back then. They are extremely rare, but these are the sorts of criteria that we would use to decide whether or not we wanted to pursue further risk information like this.

MEMBER RAY: Marty, I understand your time constraint here. Just keep the answer short. This starts off with, "Significantly changes", blah, blah, blah, "operator action." I take it from all that's been said so far that this didn't pass that gate. It wasn't viewed as a significant change in operator action.

MR. STUTZKE: Well, the fact is we've reviewed two containment overpressure credit requests, one at Vermont Yankee, and one at Brown's Ferry in detail to see that. The question is, are there other ones out in the future that may significantly change operator actions or functional requirements?

MEMBER RAY: Well, in the case of Brown's

Ferry, it didn't meet that test. Is that right? Significantly changing operator action.

MEMBER CORRADINI: For 5 percent.

MR. STUTZKE: No.

MEMBER RAY: We're not talking about the - you're talking about the one that's already gone past.

MR. STUTZKE: Yes.

MEMBER RAY: All right.

MEMBER APOSTOLAKIS: So the issue of cavitation here would be under significantly affect your basis -

MR. STUTZKE: That's my reading.

MEMBER RAY: That was my question.

MR. STUTZKE: That's one of the trips, functional requirements, redundancy, it's this sort of thing.

MEMBER RAY: Just a side discussion I had with Mike here. I'm asking about the 20 percent upgrade when I ask about significantly change operator action.

MR. STUTZKE: Right. But remember, at this point in the process, you're trying to get into the process. You haven't confirmed yes or no that it does significantly change it. You just think well, there's a potential, and we need to do some calculations, or get the licensee to do some calculations like that.

Again, the numerical acceptance guidelines in 1.174 aren't legally binding requirements. They're not the law, they provide us a basis for reasonable assurance of adequate protection, like that.

MEMBER APOSTOLAKIS: But I thought you said that 1.174 doesn't deal with adequate protection.

MR. STUTZKE: Well, the point is, if you can demonstrate that in fact you meet all the five criteria under 174, then you have adequate protection.

MEMBER APOSTOLAKIS: Violate, you may still have it.

MR. STUTZKE: You may still have it, even if you violate it. And as I said before, one of the things that happened when the Commission set this up in their last SRM, they said we want to be notified whenever you identify special circumstances. And there's a variety of reasons for that. I mean, it's a new process and whatever, but the implication when you find special circumstances is there's something really wrong. There's something wrong in your regulatory guidance, or maybe your rule is not -- it has broad implications like that.

The other thing is, in order to reject a non-risk-informed license amendment on the basis of risk, I've got to convince all the way up to the Director of NRR. It's not done at the Branch Chief level, which is where we would accept things, so that would mean I have to convince my older Branch Chief, then his Division Manager, then up through like this. I mean, it's a serious, serious thing to reject one of these things.

MEMBER CORRADINI: And that's because -- I just want to make sure, just for clarification. And that's because they've met the deterministic regulations.

MR. STUTZKE: Yes.

MEMBER CORRADINI: They're a non-risk-informed application, which all EPUs are.

MR. STUTZKE: They're in compliance with all regulations -

MEMBER CORRADINI: Unless some bell rings and a risk calculation confirms the bell, they have met the letter of the law.

MR. STUTZKE: Right. And it's not just a small delta CDF. I mean, you'd have to be way up there in core damage frequency, or large early release frequency, something like that.

MEMBER CORRADINI: And just to drive the point home back to my original question. If the calculation of the overpressure credit necessary has to literally intersect or cross the design pressure for an alarm bell to ring on risk, maybe not even then.

CHAIR SHACK: Containment is never going to see design pressure in a risk basis.

MEMBER CORRADINI: Well, there was a need to have a very large overpressure credit for a very long period of time for the yet to be reviewed 20 percent EPU.

MR. STUTZKE: Right. And I can calculate the increase in risk as a result of requiring that overpressure credit, as compared to no pressure required at all.

MR. LOBEL: We ought to be careful of the nomenclature, I think. We're not talking about the design pressure. I think we're talking about the pressure that's calculated to be in the containment. You're talking about the pressure that's there, and the pressure that's needed being close together.

MEMBER CORRADINI: Correct. Excuse me. I apologize. I apologize.

MEMBER BANERJEE: I guess the real risk is that you don't get the overpressure that you say you will. Then the pumps cavitate, and they don't work.

MEMBER CORRADINI: But as Rich said it better, I misstated it as this pressure is changing with time, and the overpressure that I need to make this criteria, and that window narrows, unless I cross that window, there's no risk-based decision that says something -- an alarm bell rings.

MEMBER BANERJEE: The real risk is -

MR. LOBEL: But if you cross the window, it would never get to Marty anyway, because we would just tell the licensee that it's unacceptable at this point.

MEMBER CORRADINI: Fine.

MEMBER BANERJEE: But this risk analysis would take into account all the uncertainties and everything, and then get the probability of not meeting this criteria.

MEMBER CORRADINI: Not this one.

MEMBER BANERJEE: And not -- and failing the pumps. Right?

MEMBER CORRADINI: No.

MEMBER APOSTOLAKIS: This risk assessment, though, is based on the two basic assumptions that were presented earlier. The magnitude of the overpressure is relevant to the duration. He's not quantifying any of that. He's assuming that these are irrelevant. All he's looking at is whether there is overpressure. Right, Marty?

MR. STUTZKE: That's correct.

MEMBER APOSTOLAKIS: Yes. So it's not really a full uncertainty analysis. It's a piece of it.

MR. STUTZKE: The uncertainty of the safety margin, no.

MEMBER APOSTOLAKIS: Sorry?

MR. STUTZKE: The uncertainty of the -

MEMBER APOSTOLAKIS: You don't look at the safety margin. That's another -

MR. STUTZKE: We're not looking at the margin. That's a separate criteria.

MEMBER APOSTOLAKIS: But did you say, though, that GE is doing something, the Owner's Group, to quantify these uncertainties?

MR. LOBEL: Well, they're doing the statistical analysis approach that we call it. It's like, I don't know if you've been briefed on best-estimate LOCA. It's the same kind of thing, where you do a realistic analysis, and then you look at the uncertainties. You get an uncertainty distribution for all the important parameters, and then you do a Monte Carlo-type calculation.

MEMBER APOSTOLAKIS: But do they make the two assumptions that Marty is making? If they make, then it's not interesting. If they don't make them, then it's interesting.

MR. LOBEL: It's not a risk-analysis.

MEMBER BANERJEE: Is it like the CSAU methodology?

MR. STUTZKE: Yes.

MEMBER BANERJEE: All right.

(Simultaneous speech.)

CHAIR SHACK: Watch what you wish for, because if you get it, they'll be able to ask for even more credit.

MEMBER BANERJEE: Yes.

MEMBER APOSTOLAKIS: No, but is that what they're doing? I don't know.

MR. STUTZKE: The downside of it is this, suppose that you do a best estimate with uncertainty analysis, and you buy off on a 95-95 acceptance criteria. What that means is there's a 5 percent chance that you don't have enough overpressure. Right? So I could put that in as a basic event in the PRA and propagate it through the model.

MEMBER APOSTOLAKIS: Are we going to see that analysis?

MR. STUTZKE: The point is maybe 95-95 is not good enough, maybe you want 99-99, and I've had some limited discussions with other people in my new office, and it becomes almost prohibitive to compute it.

MEMBER APOSTOLAKIS: No, you have to look at it in the context, the bigger context of the accident.

MR. STUTZKE: That's right. It's tough.

MEMBER BANERJEE: But we do 95-95 for -

MEMBER CORRADINI: That's for you.

MEMBER BANERJEE: I mean, okay, there's a 5 percent chance it might be more, but it's not the end of the world.

MR. LOBEL: The differences in the thing we're talking about is that if you're talking about a LOCA analysis, you're talking about a few more rods exceeding the peak clad temperature. In this case, you're talking about a pump not being able to -

(Simultaneous speech.)

MEMBER BANERJEE: I don't think you're talking about necessarily core melt, because the operator clearly can do something when he sees the pump -

MR. STUTZKE: That's true. He can throttle it down.

MEMBER SIEBER: Throttling doesn't save your -

MEMBER BANERJEE: Can't he inject cold water somewhere?

MEMBER APOSTOLAKIS: Are we going to see the GE analysis here in this Committee?

MR. LOBEL: The topical report?

MEMBER APOSTOLAKIS: After you issue an SER.

MR. LOBEL: Yes. We committed to come back to you after we issue an SER. There was a pre-brief before we started our review that GE came in and gave a pre-brief. The topical report is available. It's proprietary, but that's available if you want to look at the topical report.

MEMBER APOSTOLAKIS: But this Committee will actually have a chance to review it.

MEMBER BANERJEE: But the 99-99 is prohibitive for LOCA, because it's a very elaborate calculation, where this is going to be a much, much smaller -

MR. STUTZKE: You're well outside my

expertise.

MEMBER BANERJEE: Yes, this would be a much simpler calculation. I don't know.

MR. STUTZKE: And the containment computation is easy to -

MEMBER BANERJEE: Much easier.

MR. STUTZKE: The suction side, and I don't know how fast.

MEMBER APOSTOLAKIS: Marty, if I wanted to develop an event tree to consider various possibilities for human intervention, if I still made the two assumptions, it doesn't help me. Right? If I made the assumption of magnitude and time are irrelevant -

MR. STUTZKE: Right.

MEMBER APOSTOLAKIS: -- it wouldn't really matter. But if I wanted to do that, I would have to relax those assumptions. Correct?

MR. STUTZKE: Well, that's absolutely right. The reason why we said time wasn't important, it was a matter of looking at the fails-on-demands sorts of probabilities, failure to achieve isolation, as compared to the failure rate, the containment leakage rates, which seem to be considerably lower, even over 24 hours, 72 hours.

MEMBER RAY: Marty, it's surely the case that the uncertainty in terms of the loss of pressure due to heat transfer is much greater than the loss of pressure due to containment leakage. Now, the argument well, that's okay, because the water you're pumping is actually going to be colder at the same time, that's one that I think is worth exploring, and looking at. But my God Almighty, the pressure uncertainty has got to be very large over a long time, especially.

MR. STUTZKE: Well, that's true, but those sorts of design-related issues aren't things that are normally treated in PRA space. We worry about things failing to start, or turning off, or operators doing their -

MEMBER RAY: Okay. But I think one of our jobs is to try and realize when the process fails to include something that's important.

CHAIR SHACK: Let me just come back to the question of when I'm going to get a risk analysis. You made the argument when this was originally set up, there was an exception for EPUs. Now you're telling me that it's going to go back, and I can only get it if I'm willing to go to the Director of NRR.

MR. LOBEL: No, what we're talking about

is non-EPU amendments that deal with overpressure.

MEMBER CORRADINI: Okay.

MR. LOBEL: EPUs are -

MEMBER BANERJEE: Is this topical going to address EPUs then, or it's very general?

MR. LOBEL: Topical just talks about -

MEMBER BANERJEE: Methodology.

MR. LOBEL: -- overpressure in general. And it's only for LOCA. They're not really going into the Appendix R, ATWS, and Station Blackout.

MEMBER CORRADINI: I understand now.

MEMBER APOSTOLAKIS: No, but I don't. There will be a risk assessment with EPU?

MR. STUTZKE: Yes.

MEMBER APOSTOLAKIS: Okay. That's what you said.

CHAIR SHACK: Yes, but why is isn't it going to say that the 1.82 will be revised to say that. Oh, because that covers all requests for containment overpressure, not just EPUs.

MEMBER SIEBER: Right.

MEMBER BANERJEE: For example, for PSI 191 you might have some requests.

MEMBER SIEBER: That's right.

MEMBER BANERJEE: You may. I don't know.

MEMBER SIEBER: Why don't we move on so Marty can pick up his kids.

CHAIR SHACK: And just while Rich is here, we've talked about meeting deterministic requirements, the risk requirements. I just -- defense-in-depth is where the ACRS originally started out here. And, in fact, if you go all the way back to Reg Guide 1.1, it was issued because of an ACRS concern about independence of barriers. I mean, the whole thing was up initially to preserve independence of barriers, and we've given that up. And now we're ready to give it up to an even greater and greater extent. And the Staff agreed with us back in 1970.

MEMBER APOSTOLAKIS: Hopefully, not with us.

CHAIR SHACK: Our glorious ancestors.

MR. STUTZKE: It's very interesting. I tried over the summer to read a lot of regulatory history, and I would recommend you read Dr. Okrent's book on the history of the ACRS, because he mentions overpressure credits, specifically mentions it.

CHAIR SHACK: Positively or negatively?

MR. STUTZKE: And he said the Committee had a, "Philosophical safety concern". Okay? And I read that and went wow, they knew back then. And the resolution of that was issuance of Safety Guide 1.

CHAIR SHACK: I have more than a philosophical concern, because we've now seen events where we've used up some of this margin we thought we had with sump blockage and things like that. So giving up independence and margin still strikes me as an objectionable thing to be doing.

MR. STUTZKE: Well, the other thing I'll point out in both Dr. Okrent's book, and the second book from the NRC historian, you will not find the phrase "defense-in-depth." I found that remarkable, absolutely remarkable.

MEMBER APOSTOLAKIS: Well, it was defined in 1990.

CHAIR SHACK: Well, we'll be talking to the defense-in-depth folks tomorrow.

(Off the record comments.)

CHAIR SHACK: Do we have further comments or questions?

MEMBER APOSTOLAKIS: Well, did you conclude that all -

CHAIR SHACK: I want to make sure that you do get -

MEMBER BROWN: The local dummy is speaking right now, so if you go through these -- you started off the initial discussion with the accident occurs. The operators follow their procedures. They initiate cooling, they initiate the drywell cooling, turn it all on. And then they just -- the plant does what it does. I'm paraphrasing, or I hope I'm paraphrasing accurately. And then at some point, I presume you cool, and all the analyses say that you will not exceed containment pressures, you will not -- you'll be able to cool the core based on the analyses done, with whatever assumptions are there. Is that a correct statement for the design-basis accident?

MR. LOBEL: There is a lot of -- there's different analyses. There's analysis for peak pressure, and there's an analysis for cooling the plant. And then there is this NPSH analysis.

MEMBER BROWN: I'm assuming -- I want to deal with NP.

MR. LOBEL: Okay.

MEMBER BROWN: I'm just fundamentally -- I mean, if you had everything else in place, your pumps would work and you would cool the plant.

MR. LOBEL: Right.

MEMBER BROWN: Now, we find we have to do some things to insure that the pumps will operate under this particular scenario, and there may be even others with the sump screens, or whatever else comes up. If the operator in this circumstance here, at least the one I'm aware of, the drywell turning those coolers back off again, and continues to cool, the pressure goes up in the containment. Will it reach the design-pressure or not?

MR. LOBEL: I don't think so.

MEMBER BROWN: How far away? Is it way far away?

MR. LOBEL: In a PWR, they get closest to the design pressure at the very beginning of the accident.

MEMBER BROWN: That's fine.

MR. LOBEL: You're blowing down all the steam and water in the drywell, and it has to pass through the vent system to et to the wetwell. It can't do it all at once, because it's too small, so the pressure is going up while that stuff is trying to get down into the wetwell to get condensed.

MEMBER BROWN: I got that. I got that.

MR. LOBEL: Okay.

MEMBER BROWN: I'm going to make sure Marty gets out. So that the issue -- one of my concerns was we -- once we turn those off and he's -- the pressure doesn't exceed the containment pressure, so it's a matter of time affect on other things. But now are you setting yourself up if the operator loses control of the cool down of the reactor, and the temperatures go up for some reason, have you not put yourself closer to a circumstance where you could violate the containment?

MR. LOBEL: If the operator turns them off, and then he has to turn them back on again?

MEMBER BROWN: Some type of an error -- yes, and all of a sudden, and they don't -- all I'm worried about is moving from a regime where he doesn't have to pay any attention to it, to a regime where he makes - and if he made the mistake with the drywell coolers on, he's got margins of having something happen, whereas, if the drywell coolers are off and the pressure is up so we maintain NPSH -

MR. LOBEL: He's got a lot of margin. The peak pressure is something like 9 psig, and the design pressure is 56.

MEMBER BROWN: I'm not talking about not doing it because of NPSH, but he turns off the pumps for some reason, or he loses track of the containment pressure because he's -- whatever it is, because he's throttled the pumps.

MR. DENNIG: I don't think we had concern about them not being turned back on again.

MEMBER BROWN: I just wanted to -

MR. DENNIG: I don't believe that that was one of our concerns.

MR. LOBEL: Well, if he turned them back on, and he went -- he could still go down below the pressure.

MR. DENNIG: I mean, for a reason, for a reason of managing the accident other than the original issue.

MR. LOBEL: I mean, you can make that kind of a statement about all the accident analysis. What if the operator did something that makes things worse.

MEMBER BROWN: Well, the point is it's required now in this circumstance to do this, in order to maintain the NPSH. And we ended up, and maybe this is mindless, but that finish circumstance where they added the power uprates, and now they had this other system that used to not be required, now it was required. Not only was it required, but it required a specific pump coast - some type of a coast down in order to maintain accident margins, and then something went wrong once they got into this, and they ended up having -- they were short, and they could have the -- they were at low power instead of high power when the problem occurred. You knew you gradually you increase the power rating, and you trim things off, taking advantage of margins. It's similar to Bill's comment about -

MR. LOBEL: These fan coolers are non-safety equipment, if they -

MEMBER BROWN: Well, these other ones were, too, initially.

MR. LOBEL: In other BWRs, they don't continue into the accident. They're not loads that are picked up, so they don't operate at all. In this case, they do, so the operator has to turn it off. He has two hours to do it, and he has three different ways he can do it from the control room, and outside the control room. The drywell coolers, so it's -

MEMBER BANERJEE: I have a quick question. When is this GE report available?

MR. LOBEL: It's available now. It's in ADAMS.

MEMBER BANERJEE: Would it be in time for Brown's Ferry?

MR. LOBEL: No, it probably won't be done before March. It may be, but I don't -

MR. RULAND: They haven't referred it in their submittal. The Staff has already issued the SE for the Brown's Ferry uprate, and it was - we did our review at 120 percent. Right, Rich?

MR. LOBEL: We did our review at 120 -

MR. RULAND: Well, this isn't going to affect Brown's Ferry.

MEMBER BANERJEE: This is just a general -

MR. LOBEL: This is a general topical report.

MR. RULAND: Right.

MEMBER APOSTOLAKIS: Can you go to slide 6? It seems to me from what I've heard in Rich's presentation and all the discussion, the real principle that applies here is number 3, maintain sufficient safety margins. Also, 2, but 3 is the key one. All the discussion here was 3, and 3 is not dealt with at all. Is that correct, Marty?

MR. STUTZKE: Well, all of the principles are dealt with, but you're asking about how we divvy up the work.

MEMBER APOSTOLAKIS: Well, I mean, is there any calculation to demonstrate that sufficient safety margins are maintained? I don't believe so, because you are dealing only with the existence of overpressure. You have made those two assumptions, that the magnitude doesn't matter, and the duration doesn't matter.

MR. STUTZKE: Okay.

MEMBER APOSTOLAKIS: And you have eliminated 3. All the discussion is about that.

MEMBER SIEBER: You want actual to equal required. For an actual pressure to equal required pressure.

MEMBER APOSTOLAKIS: Yes.

MEMBER SIEBER: That's margin. There's margin built into that.

MEMBER APOSTOLAKIS: But it -

MEMBER SIEBER: 3 percent.

MEMBER APOSTOLAKIS: But they don't quantify that.

MEMBER SIEBER: 3 percent.

MEMBER APOSTOLAKIS: No, no, no. These are not -

MR. STUTZKE: It's not quantified, but yes, it goes back to how their work is divvied up. Really, the PRA guys do number 4. And everything else is a problem.

MEMBER APOSTOLAKIS: Is problem. But you do quantify 3, or maybe GE does. Anyway, I just -

MR. STUTZKE: Yes. That's the intention.

MEMBER APOSTOLAKIS: That's the intention. Right.

MEMBER ABDEL-KHALIK: Have you at any time sort of really addressed the underlying issue that Howard brought up about over-cooling to the containment, and the fact that as the pressure increases, that the rate of change of saturation pressure with saturation temperature actually increases? And, therefore, small changes in temperature can cause large changes in pressure.

MR. LOBEL: Which is why the suppression pool temperature is so much more important than the pressure above the water level.

MEMBER ABDEL-KHALIK: In the other direction, meaning that if you have extra heat transfer to the containment walls, for example, so that the overall temperature is reduced, that will cause significant changes in containment pressure so that you may not have enough containment pressure to give you adequate NPSH. And the problem becomes worse as the temperature increases, or as the pressure increases.

MR. LOBEL: I don't know of a calculation like that, but I would guess that -

MEMBER ABDEL-KHALIK: Just look at the

steam table. The rate of change of saturation temperature with saturation pressure.

MR. LOBEL: Right. That's what we were talking about with why the temperature is so much more important going up.

MEMBER RAY: But you're assuming equilibrium, which I -- if we did the analysis, I would -- that's exactly where I would go. That's the question that Said is asking. The water stays hot, but the containment atmosphere is cooled more than you think, and now your assumption that well, the water will always be cooled at the same time that the atmosphere is being cooled and depressurized by the heat transfer, that isn't necessarily a correct assumption, it doesn't seem to me.

MEMBER SIEBER: It would probably go to saturation.

MEMBER RAY: Well, I don't know, Jack. I can imagine hot water -

MEMBER SIEBER: That's as hot as it can get.

MEMBER RAY: -- and cool containment temperature. Well, then let's look at the picture.

MR. LOBEL: Well, it can't be too cool because what's heating it, the suppression pool is hot. There's the vapor -

MEMBER RAY: Just cooling off is all we're talking about. Look, this is just a matter of quantifying something. That's all.

MR. LOBEL: The vapor is going to be in equilibrium with water most likely.

MEMBER RAY: Is that enough?

MR. LOBEL: And the gas in the wetwell is in contact with the vapor, so it's hard to see that it's going to get too much -

MEMBER ABDEL-KHALIK: Too much time constant of the containment shield wall, the thick concrete. What's the time constant of that?

MR. LOBEL: Of the concrete?

MEMBER ABDEL-KHALIK: Right.

MR. LOBEL: Well, that the torus is metal.

MEMBER ABDEL-KHALIK: But there are other means of heat loss to the world.

MEMBER RAY: It's just a matter it needs to be shown rather than -

CHAIR SHACK: We have -- just to come back to defend a little bit. Whenever we've seen calculations that relax the conservative assumptions that Rich has talked about, we've always found that the net has gone down. I mean, I can't say that we've looked at every possible combination -

MEMBER RAY: The net margin?

CHAIR SHACK: The net required, the required NPSH has always gone down.

MR. LOBEL: You're saying you're going to have a lot of heat transfer from the wet well atmosphere, but the water is going to stay hot. I don't know of a calculation like that. I have a feeling it would be hard to get in that situation, but I don't -

MEMBER RAY: I'm not trying to make an argument. I'm only try to illustrate where it seems to me there's uncertainty. And, to me, containment leakage is a small player, containment cooling is a big player by comparison. And the argument that oh, well, the water will cool down more rapidly than the loss of pressure due to containment cooling; well, maybe so, but let's see some numbers.

MR. LOBEL: But the containment cooling is mostly from the heat exchanger, which is in the water.

MEMBER RAY: I don't know that. There's heat transfer that takes place over many hours here. This is not something that I can intuit the way perhaps you've been able to do.

MR. LOBEL: Well, I don't know the -- I

can't say I can point to a calculation that we've done, or GE's done that way. We might be able to do one.

MEMBER RAY: When you're down to the kind of margins we're talking about here, you've got to do it, it seems to me.

MEMBER SIEBER: Well, those kinds of calculations have been done typically by designers of containment where they look at the material mass, thermal energy, heat transfer.

MEMBER RAY: Yes. I would replace the -

MEMBER SIEBER: That part of the information is available.

MEMBER RAY: The pumps at my plant, and we sharpened every pencil there was, but that's what happened. So I've seen this done before, and I'm just not that sanguine about these arguments.

MEMBER SIEBER: Yes, heat loss that way is important.

CHAIR SHACK: Marty is about to get in trouble here, so unless you have a burning question.

MEMBER SIEBER: Yes. Just stand up and say goodbye.

MR. STUTZKE: Thank you.

(Simultaneous speech.)

MEMBER MAYNARD: I really appreciate the discussion, and I think Rich and Marty did a good job. They provided information that gives me something to think about. I'm not sure it changes my overall opinion, but it certainly shed light on some aspects of it, and the regulatory process that I didn't know. And I think they've done a good job of fielding our questions and maintaining -

CHAIR SHACK: Well said, well said.

MEMBER MAYNARD: So I appreciate it.

MEMBER APOSTOLAKIS: Do we have any conclusions from today's meeting?

CHAIR SHACK: Well, we get to discuss that. When is -

MEMBER STETKAR: You mentioned you've made changes to the White Paper. Are we going to -- when is the Rev.1?

MR. RULAND: Based on -- having listened to this discussion, it kind of played out the way I thought it would play out. What we would very likely do at this juncture is revise the White Paper, and I suspect we will send a Commission paper, and the White Paper will be attached to the Commission paper. And that's how we likely will proceed. And the Commission has directed us to send policy papers, if needed.

And, like I said, I didn't hear that you were overjoyed by our analysis, and we were in violent agreement. I heard that you basically have viewed this -- while I think we enlightened you with some kind of our argument in one place, I don't hear that you've moved much in your view on the matter. So once we prepare a Commission paper, you'll see the White Paper as an attachment, very likely.

Now, that's not a commitment, but that's -- based on what I heard today, I've got to talk to my boss here, of course, Jack Rowe, who is sitting to my left, and I invited Jack, and Jack wanted to come here because he wanted to hear these discussions first-hand. And I would -- anyway, thank you for the discussions, because I think they really mirrored the discussions and the debates we've had all along in this matter. So we'll -- as I said in the beginning, we want to drive this issue to closure.

We can't -- these can be entertaining, but we want to really drive this -- we've got to make a decision in this matter, frankly, at the Commission level, and move forward. And I think you can appreciate that. Just as a matter of how we conduct our business. Anyway, so that's maybe -- hopefully I answered your question.

MEMBER STETKAR: Yes, thanks.

MR. ROWE: George asked a question, and I think the answer is that your position that you had presented to the Commission hasn't changed.

CHAIR SHACK: No, I think we need to discuss things.

MR. ROWE: Okay.

CHAIR SHACK: I mean, this is all -

MR. ROWE: I'm interested in the results of these conversations.

CHAIR SHACK: The results of those conversations -

MR. ROWE: How would we get those absent a letter?

CHAIR SHACK: We'll have to discuss that.

MR. RULAND: And any intelligence, any information you can give us just beyond us listening to this meeting, would be greatly appreciated. But, like I said, we're already -- my direction to our staff is write the paper right now. And that's going to be our direction, because we really can't wait. We've got to move forward.

MEMBER BANERJEE: You've got a new piece of information in the topical. Right? Which might actually be very useful.

MR. LOBEL: I really don't think the topical is going to help with any of the issues you're talking about here. It's really just one way of doing the calculations. And, in fact, the way the Owner's Group posed the topical is they want to use the topical to justify the deterministic analysis. They don't even want the statistical analysis to be the licensing basis. And we're talking with them about that, but -

MR. DENNIG: The statistical analysis is just a randomization of initial conditions, and then doing calculations.

MEMBER BANERJEE: Then it's not a CSAU.

MR. LOBEL: It's close to a CSAU. They don't want to call it a CSAU, and they didn't do it exactly like a CSAU. But the questions the members have been asking here aren't answered in the topical. It isn't going to help.

MS. ABDULLAHI: Can I make a point. This is Zeyna Abdullahi. We had this topical report briefing in February `08, and this is the case where the Staff was saying that they have best-estimate -- no, that they have very conservative LOCA containment calculations. And the BWROG said that, and the ACRS was also saying there's not a consistency from plant to plant. And the BWROG proposed with the help of the Staff a statistical approach for the LOCA containment analysis with uncertainty, treatment as opposed to very conservative, like Rich was saying. And we had that briefing, and you will review it later on. But the request in the past of having a LOCA with uncertainty so you have more confidence in the calculation without being overly conservative, I think that's where the topical report provides some help.

CHAIR SHACK: Again, let me express our thanks. It's been a very interesting discussion, done very well, and we will be talking.

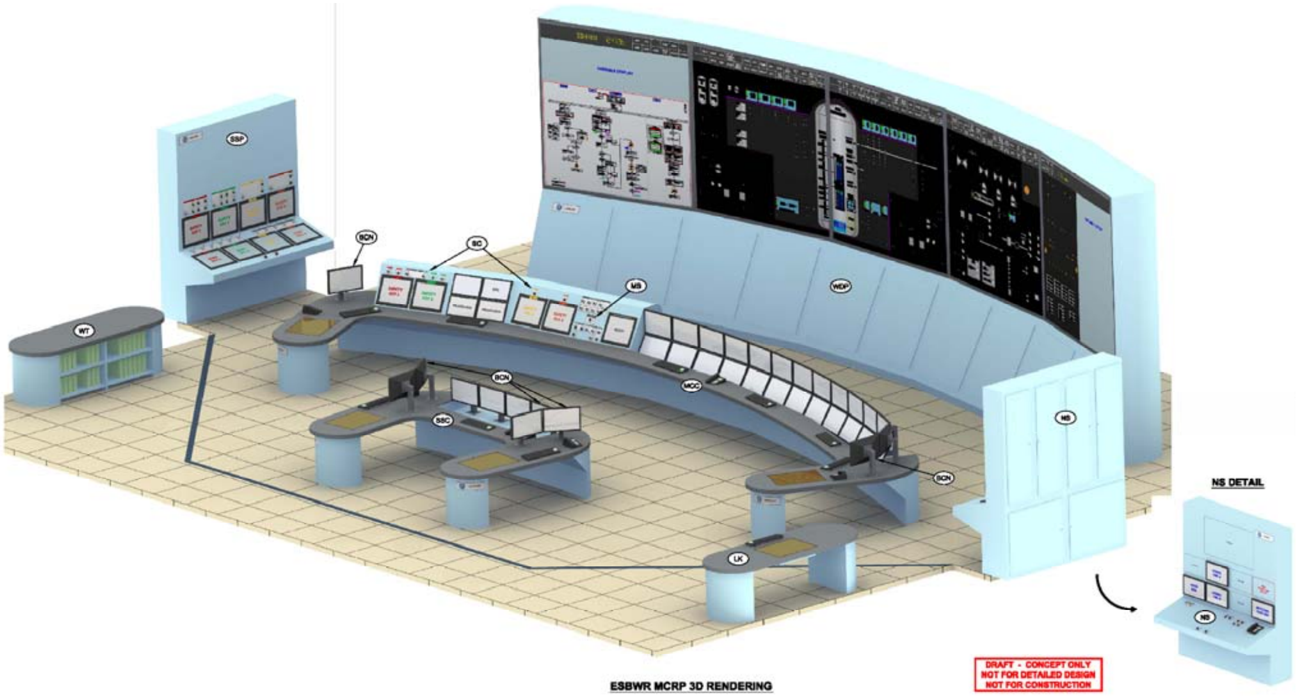
MR. RULAND: Rich and Marty, thank you very much.

CHAIR SHACK: We'll take a 15-minute break.

(Whereupon, the proceedings went off the record at 5:20 p.m.)

ESBWR DCD Chapter 7 DCIS Overview

Full/Subcommittee ACRS Meeting



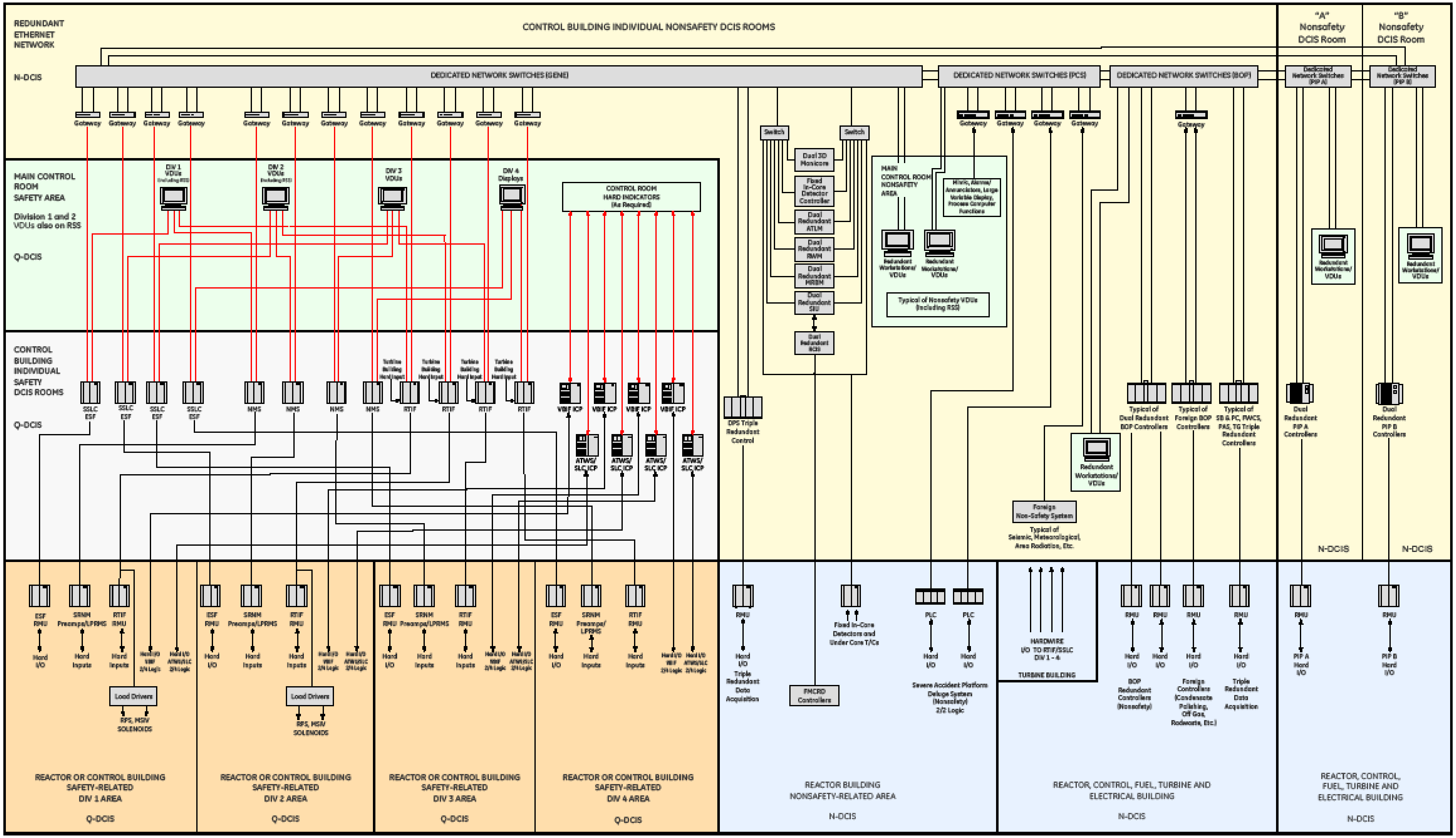
ESBWR MCRP 3D RENDERING

Rich Miller
Ira Poppel
Steve Kimura
Dec 3-4, 2008

HITACHI

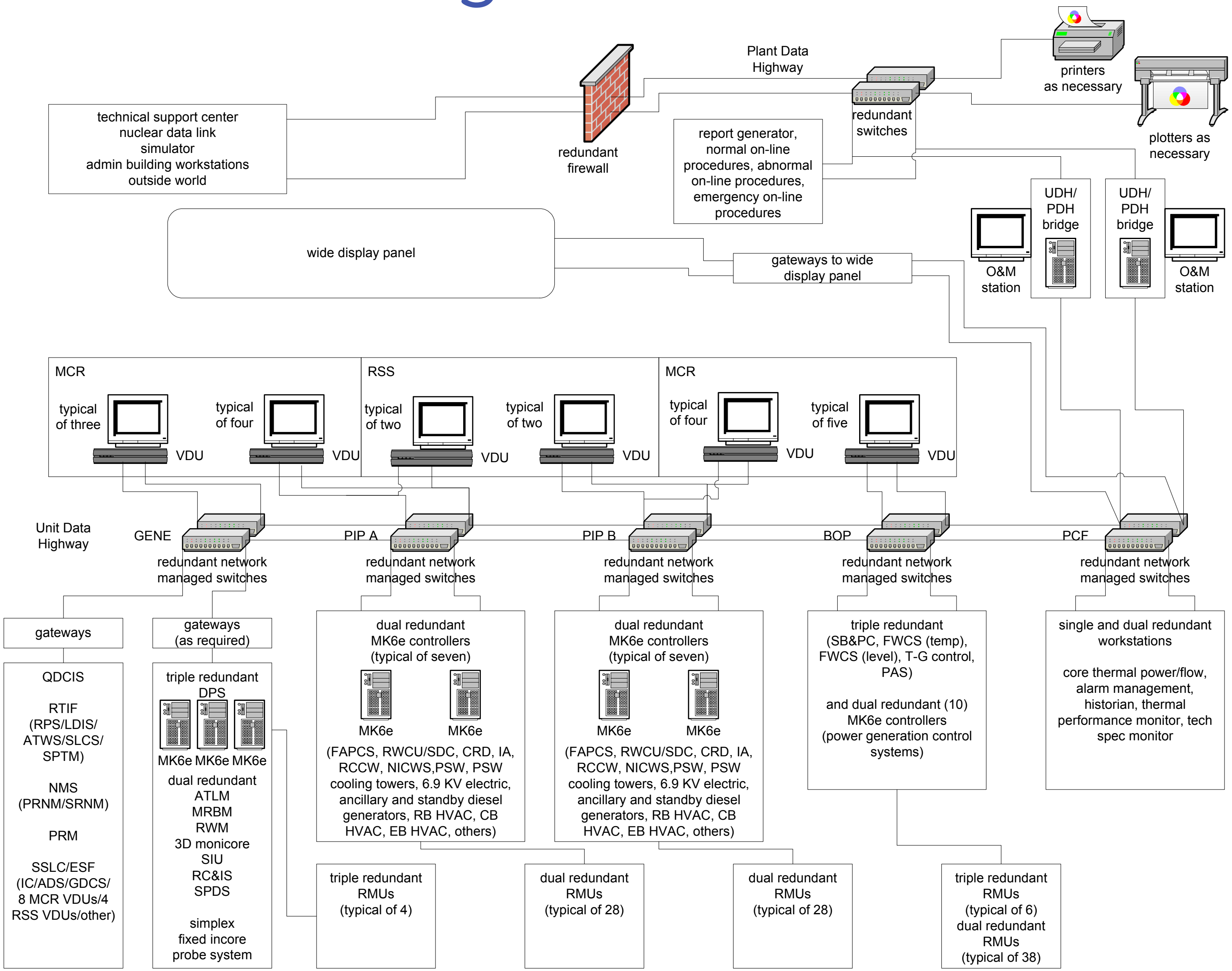
ESBWR DCIS

ESBWR Distributed Control and Information System (DCIS) Functional Network Diagram



HITACHI

ESBWR DCIS Organization - Continued



technical support center
nuclear data link
simulator
admin building workstations
outside world

redundant firewall

Plant Data Highway

redundant switches

printers as necessary

plotters as necessary

report generator, normal on-line procedures, abnormal on-line procedures, emergency on-line procedures

wide display panel

gateways to wide display panel

UDH/PDH bridge

UDH/PDH bridge

O&M station

O&M station

MCR

typical of three VDU

typical of four VDU

RSS

typical of two VDU

typical of two VDU

MCR

typical of four VDU

typical of five VDU

Unit Data Highway

GENE redundant network managed switches

PIP A redundant network managed switches

PIP B redundant network managed switches

BOP redundant network managed switches

PCF redundant network managed switches

gateways

QDCIS

RTIF
(RPS/LDIS/
ATWS/SLCS/
SPTM)

NMS
(PRNM/SRNM)

PRM

SSLC/ESF
(IC/ADS/GDCS/
8 MCR VDUs/4
RSS VDUs/other)

gateways
(as required)

triple redundant DPS

MK6e MK6e MK6e

dual redundant
ATLM
MRBM
RWM
3D monicore
SIU
RC&IS
SPDS

simplex
fixed incore
probe system

dual redundant MK6e controllers (typical of seven)

MK6e MK6e

(FAPCS, RWCU/SDC, CRD, IA, RCCW, NICWS,PSW, PSW cooling towers, 6.9 KV electric, ancillary and standby diesel generators, RB HVAC, CB HVAC, EB HVAC, others)

dual redundant MK6e controllers (typical of seven)

MK6e MK6e

(FAPCS, RWCU/SDC, CRD, IA, RCCW, NICWS,PSW, PSW cooling towers, 6.9 KV electric, ancillary and standby diesel generators, RB HVAC, CB HVAC, EB HVAC, others)

triple redundant (SB&PC, FWCS (temp), FWCS (level), T-G control, PAS)

and dual redundant (10) MK6e controllers (power generation control systems)

single and dual redundant workstations

core thermal power/flow, alarm management, historian, thermal performance monitor, tech spec monitor

triple redundant RMUs (typical of 4)

dual redundant RMUs (typical of 28)

dual redundant RMUs (typical of 28)

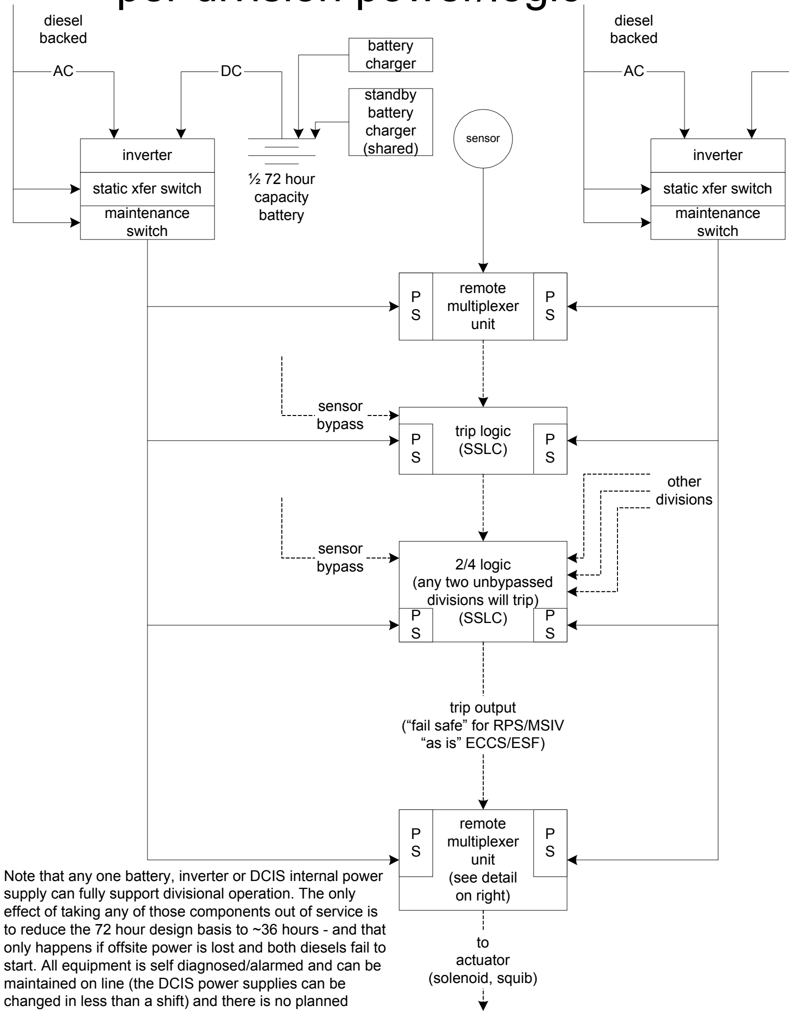
triple redundant RMUs (typical of 6) dual redundant RMUs (typical of 38)

Q-DCIS

- Q-DCIS organized into
  - RTIF/NMS (reactor trip system)
  - SSLC/ESF (ECCS and information systems)
  - ATWS/SLC and VBIF
- Q-DCIS is deterministic
- Q-DCIS has four divisions
- Q-DCIS is N-2
- RTIF/NMS and SSLC/ESF functions implemented on diverse hardware/software platforms
- Q-DCIS is physically, electrically and data isolated between divisions and between Q-DCIS and N-DCIS

Q-DCIS Power

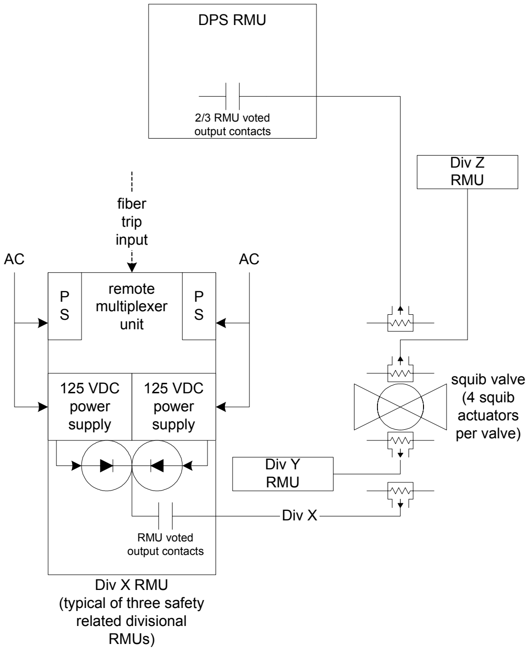
per division power/logic



Note that any one battery, inverter or DCIS internal power supply can fully support divisional operation. The only effect of taking any of those components out of service is to reduce the 72 hour design basis to ~36 hours - and that only happens if offsite power is lost and both diesels fail to start. All equipment is self diagnosed/alarmed and can be maintained on line (the DCIS power supplies can be changed in less than a shift) and there is no planned requirement to ever make a division completely “black”.

per actuator

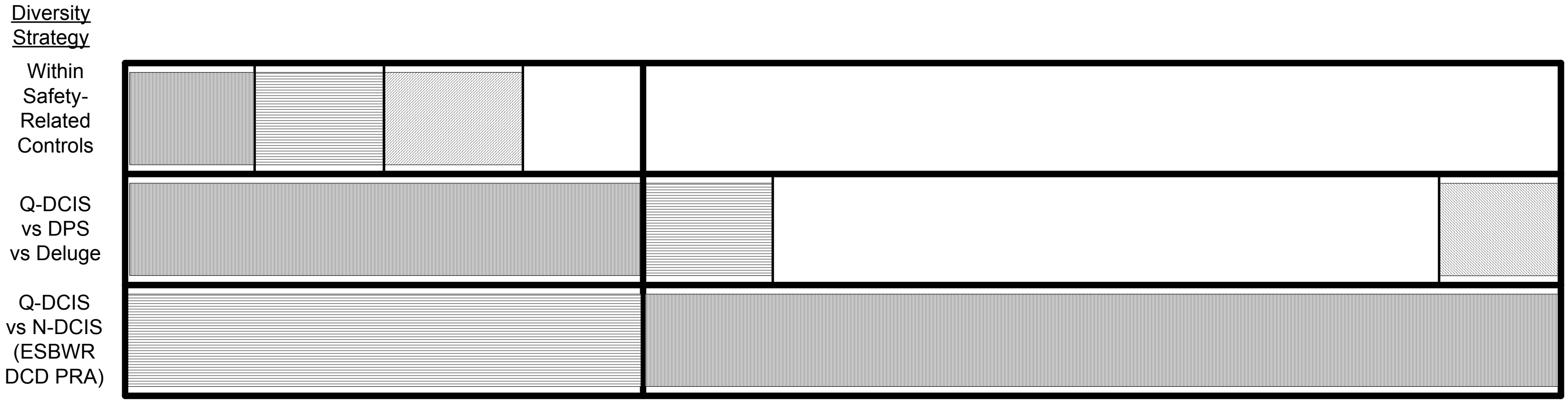
Typical of SRV solenoids, DPV valves, GDCS valves, equalizing valves; any one (of three) squibs (divisions) or DPS will open the valve.
Lines with two isolation valves in series use two solenoids per valve; lines with one isolation valve in series with a check valve use four solenoids.
Per division, any one of two AC inputs (inverters/batteries), allows the squib to fire.
Overall scheme allows any division to be taken out of service, incur a DBA and still accept a single additional failure and still open all valves.



Div X RMU
(typical of three safety related divisional RMUs)

ESBWR DCIS Overall Diversity

Safety Category	**Safety-Related**				**Nonsafety-Related**						
	Q-DCIS				**N-DCIS**						
Platform/ Network Segment	**RTIF NMS**	**SSLC/ ESF**	**Independent Control platform**	**other**	**GENE**		**PIP A/B**	**BOP**		**PCF**	
architecture	**divisional**	**divisional**	**divisional**	**note 1**	**Triple Redundant (DPS)**	**Dual Redundant**	**Dual Redundant**	**Triple Redundant**	**Dual Redundant**	**Workstations**	**PLC (Deluge)**



Note 1 – RSS provides operator workstations at appropriate diverse locations outside the main control room in accordance with GDC 19. See DCD section 7.1.3.2.3.2

Note 2 – Crosshatching denotes different platforms or networks

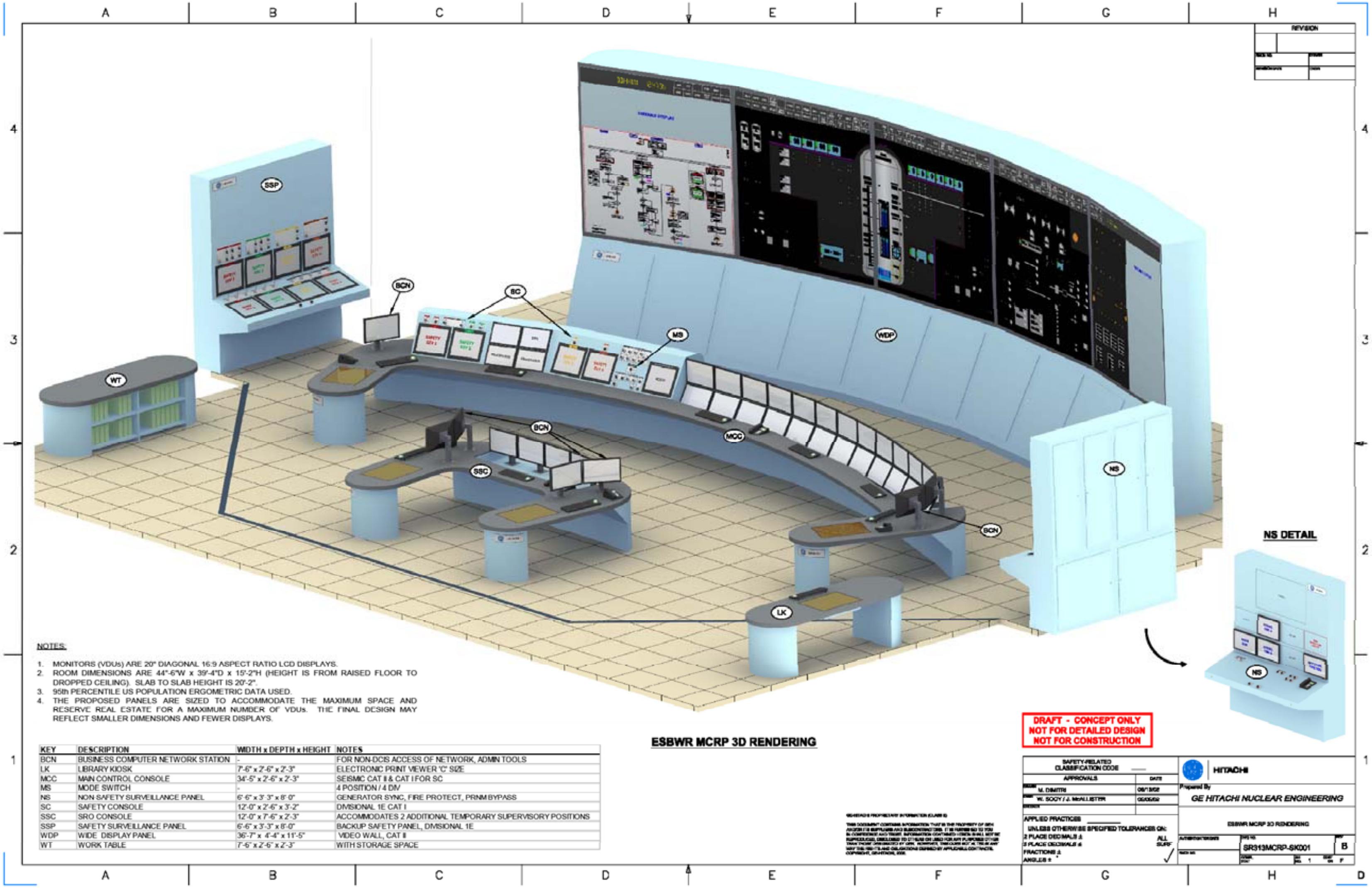
N-DCIS

- N-DCIS is organized into five independent dual redundant network segments
  - GENE (contains DPS)
  - PIP A (investment protection/RTNSS)
  - PIP B (investment protection/RTNSS)
  - BOP (power generation)
  - Plant Computer Functions
- A/B N-DCIS components located in separate rooms/fire zones
- N-DCIS components dual or triply redundant powered by two or three uninterruptible power systems
- Important reactor control systems segmented
- Networks are not used for closed loop control
- N-DCIS components diverse from Q-DCIS components

ESBWR Diverse Protection System

- Provides manual and automatic
  - Backup scram functions
    - (Rx level, Rx pressure, pool temperature, drywell pressure)
  - Backup MSIV isolation functions
    - (Rx steam flow, Rx level)
  - backup ADS and GDCS initiation
  - Backup IC initiation
  - Backup process isolation functions
  - SLCS initiation
- Mitigates loss of feedwater heating (SRI, SCRRI)
- Initiates ARI, SRI/SCRRI, all control rod run-in
- Initiates FW runback
- Initiates level 9 FW pump trip

ESBWR Main Control Room



NOTES:

1. MONITORS (VDUs) ARE 20" DIAGONAL 16:9 ASPECT RATIO LCD DISPLAYS.
2. ROOM DIMENSIONS ARE 44'-6"W x 39'-4"D x 15'-2"H (HEIGHT IS FROM RAISED FLOOR TO DROPPED CEILING). SLAB TO SLAB HEIGHT IS 20'-2".
3. 95th PERCENTILE US POPULATION ERGOMETRIC DATA USED.
4. THE PROPOSED PANELS ARE SIZED TO ACCOMMODATE THE MAXIMUM SPACE AND RESERVE REAL ESTATE FOR A MAXIMUM NUMBER OF VDUs. THE FINAL DESIGN MAY REFLECT SMALLER DIMENSIONS AND FEWER DISPLAYS.

KEY	DESCRIPTION	WIDTH x DEPTH x HEIGHT	NOTES
BCN	BUSINESS COMPUTER NETWORK STATION	-	FOR NON-DCIS ACCESS OF NETWORK, ADMIN TOOLS
LK	LIBRARY KIOSK	7'-6" x 2'-6" x 2'-3"	ELECTRONIC PRINT VIEWER 'C' SIZE
MCC	MAIN CONTROL CONSOLE	34'-5" x 2'-6" x 2'-3"	SEISMIC CAT II & CAT I FOR SC
MS	MODE SWITCH	-	4 POSITION / 4 DIV
NS	NON-SAFETY SURVEILLANCE PANEL	6'-6" x 3'-3" x 8'-0"	GENERATOR SYNC, FIRE PROTECT, PRNM BYPASS
SC	SAFETY CONSOLE	12'-0" x 2'-6" x 3'-2"	DIVISIONAL 1E CAT I
SSC	SRO CONSOLE	12'-0" x 7'-6" x 2'-3"	ACCOMMODATES 2 ADDITIONAL TEMPORARY SUPERVISORY POSITIONS
SSP	SAFETY SURVEILLANCE PANEL	6'-6" x 3'-3" x 8'-0"	BACKUP SAFETY PANEL, DIVISIONAL 1E
WDP	WIDE DISPLAY PANEL	36'-7" x 4'-4" x 11'-5"	VIDEO WALL, CAT II
WT	WORK TABLE	7'-6" x 2'-6" x 2'-3"	WITH STORAGE SPACE

ESBWR MCRP 3D RENDERING

DRAFT - CONCEPT ONLY
NOT FOR DETAILED DESIGN
NOT FOR CONSTRUCTION

ESBWR Remote Shutdown System (RSS)

- ESBWR RSS not really a “system” – instead two auxiliary control rooms with RSS panels located in Div 1 and Div 2 quadrants of the Reactor Building
- GDC 19 RSS requirements are met by the manual scram and isolation switches on the panels
- With offsite power available, either RSS panel can operate BOP normally for plant shutdown
- With only diesel power available, either RSS panel can operate PIP A or PIP B systems for plant shutdown
- With only safety-related batteries available, either RSS panel can operate division 1 or division 2 systems for plant shutdown

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Presentation to the 558th ACRS Meeting

Summary of Staff Review of ESBWR Design Certification Document Chapter 14 and Tier 1

Presented by Eric Oesterle
Lead Project Manager (NRO/DNRL)
December 4, 2008

Staff Review of ESBWR Chapter 14 and Tier 1
Overview of Design Certification

Purpose

- Provide an update of the status of the staff's review of ESBWR DCD Tier 2, Chapter 14, Initial Test Program and ITAAC, and Tier 1, since the 557th ACRS Full Committee meeting

Staff Review of ESBWR Chapter 14 and Tier 1 Overview of Design Certification

Regulations:

- 10 CFR 50.34(b)(6)(iii) and 10 CFR 52.79(a)(28) - Initial Test Program
- 10 CFR 53.27(b)(1) - ITAAC

Regulatory Guidance:

- Standard Review Plan 14.2, Initial Plant Test Program
- Standard Review Plan 14.3, Inspections, Tests, Analyses and Acceptance Criteria (ITAAC)
- Reg. Guide 1.68, Initial Test Programs for Water-Cooled Nuclear Power Plants
- Reg. Guide 1.20, Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing
- Reg. Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)
- Reg. Guide 1.206, Combined License (COL) Applications for Nuclear Power Plants

Staff Review of ESBWR Chapter 14 and Tier 1 Overview of Design Certification

Summary of Staff Review of ESBWR Chapter 14 and Tier 1:

- RAIs issued: 539
- RAI responses submitted: 509
- RAIs resolved: 476

Summary of Staff Review of ESBWR Section 14.2, Initial Test Program:

- RAIs issued: 99 (1 new RAI since 557th ACRS mtg)
- RAIs resolved: 93
- Unresolved RAIs associated with:
  - expansion, vibration and dynamic effects testing
  - testing of digital instrumentation and control system functions
  - safety system logic and control pre-operational testing
  - lead detection and isolation system pre-operational testing
  - reactor internals vibration testing
  - AC power distribution system pre-operational testing
  - incomplete description of pre-operational testing for DCIS

Staff Review of ESBWR Chapter 14 and Tier 1 Overview of Design Certification

Summary of Staff Review of ESBWR Tier 1 and Section 14.3, ITAAC:

- RAIs issued: 440 (3 new RAIs since 557th ACRS mtg)
- RAIs resolved: 383 (21 RAIs resolved since 557th ACRS mtg)

- Unresolved RAIs associated with:
  - tables of key aspects, analyses, and design features included in ITAAC
  - interface materials (offsite power and plant service water system)
  - digital instrumentation and control systems
  - human factors engineering
  - electrical systems
  - containment systems
  - reactor systems
  - format and consistency issues across similar ITAAC
  - security design features

Staff Review of ESBWR Chapter 14 and Tier 1 Overview of Design Certification

Summary

- NRO staff continues to engage with GEH to obtain satisfactory resolutions of open items associated with review of the Initial Test Program and ITAAC that are necessary to develop the staff's Final Safety Evaluation Report (FSER) for Tier 1 and Chapter 14 of the ESBWR Design Certification Document

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Presentation to the ACRS Full Committee

ESBWR Design Certification Review
Chapter 7,“Instrumentation and Controls”

December 4, 2008

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 7

Purpose

- Brief the Subcommittee on the staff's continuing review of the ESBWR DCD Application Sections
  - 7.1 "Introduction"
    - Software Development Activities
    - Diversity and Defense-in-Depth Assessment
    - Setpoint Methodology
    - Data Communication Systems
  - 7.2 "Reactor Trip Systems"
  - 7.3 "Engineered Safety Features Systems"
  - 7.4 "Safe Shutdown Systems"
  - 7.5 "Information Systems Important to Safety"
  - 7.6 "Interlock Systems"
  - 7.7 "Control Systems"
  - 7.8 "Diverse Instrumentation and Control Systems"
- Answer the Committee's questions

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 7 Review Team

- Project Manager
  - Dennis Galvin

- Technical Reviewers
  - Hulbert Li, Lead
  - Leroy Hardin
  - Sang Rhow
  - Royce Beacom
  - Dinesh Taneja
  - Joseph Ashcraft
  - Kimberley Corp
  - Eugene Eagle
  - Thomas Fredette
  - Jack Zhao

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 7 Presentation

Outline of Presentation

- Applicable Regulations
- RAI Status Summary
- SER Technical Topics of Interest
  - Key I&C DAC/ITAAC Items
  - Key SER Open Items
- Discussion / Committee Questions

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 7

Key Regulations

- 10 CFR 50.55a(a)(1), 10 CFR 50.55a(h)(3), 10 CFR 50.34(f)(2), 10 CFR 50.62, and 10 CFR 52.47(b)(1)
- 10 CFR Part 50, Appendix A, GDC 1, 2, 4, 10, 13, 15, 16, 19, 20, 21, 22, 23, 24, 25, 28, 29, 33, 34, and 35

Principal Review Guidance

- SRP Section 7, including Branch Technical Positions
- SRP Sections 14.3 and 14.3.5
- Regulatory Guides 1.22, 1.47, 1.53, 1.62, 1.75, 1.97, 1.105, 1.118, 1.151, 1.152, 1.168, 1.169, 1.170, 1.171, 1.172, 1.173, 1.180, 1.189, 1.204, and 1.209
- SRM on SECY-93-087 and SECY-92-053

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 7

<u>RAI Status Summary: SRP Chapter 7</u>

- Original number of RAIs = 276
- Number of RAIs resolved = 206
- Number of Remaining Open Items = 70

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 7 Summary

The staff followed SRP Chapters 7 & 14 Guidance to review high level functional requirements and design commitments for:

- IEEE-603 criteria compliance
- Life-cycle design process
- Setpoint methodology
- Diversity & Defense-in-Depth
- Data Communication

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 7 Summary

RAI open items status

- Most of the remaining open items are clarification/consistency related issues
- No safety significant technical issues that need resolution

ACRS Full Committee Presentation
ESBWR Design Certification Review
Committee Questions

<u>Discussion/Committee Questions</u>

Presentation to the ACRS Full Committee

Safety Review of the
Vogtle Electric Generating Plant
Early Site Permit Application and
Limited Work Authorization Request

December 4, 2008

Purpose

- To provide the ACRS an overview of the staff’s safety review and conclusions on:
  - The Vogtle Electric Generating Plant (VEGP) Early Site Permit (ESP) Application
  - The VEGP Limited Work Authorization (LWA) Request
- Address the Full Committee’s questions

UNITED STATES NUCLEAR REGULATORY COMMISSION

Meeting Agenda

Early Site Permit Application Review:

- Remaining Schedule Milestones
- Key Review Areas / Resolution of Open Items
- Advanced Safety Evaluation Report (SER) Conclusions

Limited Work Authorization Review:

- VEGP LWA Request Summary
- Review of LWA Activities
- LWA Conclusion
- Discussion / Questions

Remaining Milestones

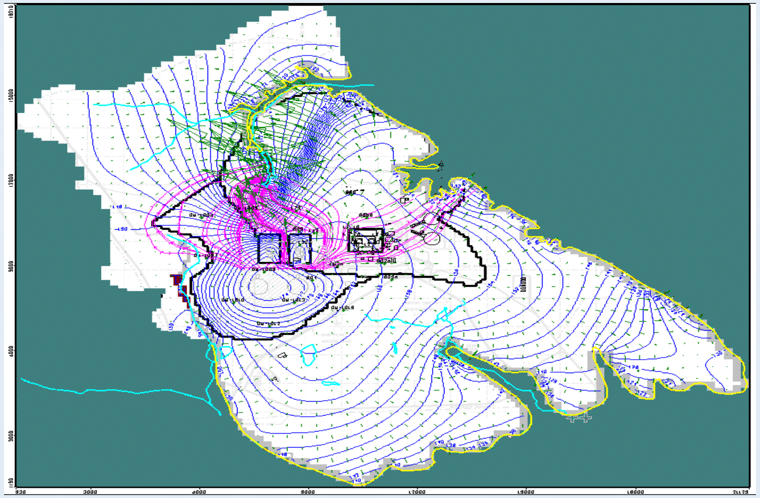
- ACRS Final Letter Assumed – 1/2009
- Final SER Issuance – 2/5/2009
- Mandatory Hearing – 3/23/2009
- Commission Decision Assumed – Summer/Fall 2009

Key Review Areas for ESP/LWA

- The staff completed its review of the following areas for the ESP:
  - 2.1 - Geography and Demography
  - 2.2 - Nearby Industrial, Transportation, and Military Facilities
  - **2.3 - Meteorology (1)**
  - **2.4 - Hydrology (4)**
  - **2.5 - Geology, Seismology, Geotechnical Engineering (22)**
  - 3.5.1.6 - Aircraft Hazards
  - 11 - Doses from Routine Liquid and Gaseous Effluent Releases
  - **13.3 - Emergency Planning (13)**
  - 13.6 - Physical Security
  - 15 - Accident Analyses
  - 17 - Quality Assurance
- Resolution of all Open Items **(Bold)** discussed in the Advanced SER

- The staff completed its review of the following areas for the LWA:
  - 2.5.4 – Stability of Subsurface Materials and Foundations
  - 3.8.5 – Foundations
  - 13.7 – Fitness For Duty Program
  - 17 – Quality Assurance Program

Section 2.4: Hydrology



Section 2.4 Hydrologic Hazard Analyses

- Floods induced by rain, dam break, hurricane, and tsunami.
- Low water impacts
- Ice impacts
- Water use impacts
- Groundwater flow and contamination transport analyses

2.4 Hydrology

- Section 2.4.8: Cooling Water Canals and Reservoirs (**OI 2.4-1**)
  - Issue: Do canals or reservoirs are used as any external water source for safety-related cooling water?
  - Resolution: Staff confirmed that safety-related cooling water is provided not from canals and reservoirs, but from groundwater wells. Based on aquifer characteristics, staff determined that the aquifer has sufficient capacity for initial filling and occasional makeup of two proposed water storage tanks - **Closed**
- Section 2.4.12: Groundwater (**OI 2.4-2**)
  - Issue: Predict future hydrogeological conditions to determine the safety of proposed facilities from groundwater-induced loadings.
  - Resolution: The applicant provided additional field hydrogeologic data (e.g., the unconfined aquifer characters, a refined recharge and hydraulic conductivity maps). NRC staff analyzed the groundwater regime with a post-construction setting and the provided data, and confirmed that a maximum water table elevation (165 ft msl) is far below the site grade (220 ft msl) - **Closed**

2.4 Hydrology (Con’t)

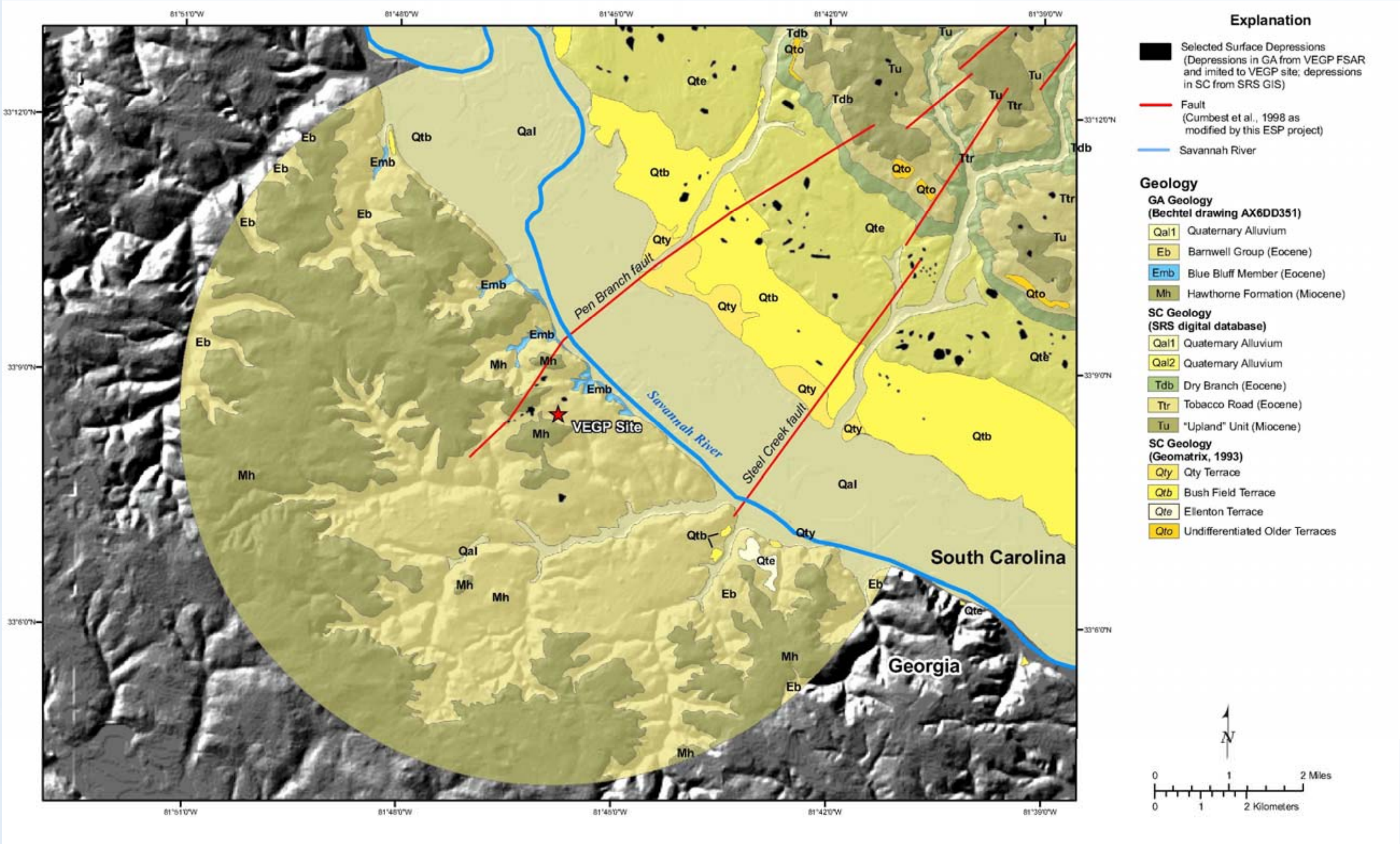
2.4.13: Accidental Releases of Radionuclides In Ground Waters

- **OI 2.4-3**
  - Issue: Consider the potential change in flow direction within the Water Table aquifer and all feasible groundwater pathways.
  - Resolution: The applicant provided additional field data; Analyses by the applicant and the NRC staff examined post-construction settings, and alternative pathways (four alternative pathways), considering an adequate number of combinations of release locations and feasible pathways - **Closed.**
- **OI 2.4-4**
  - Issue: Specify the nearest point along each potential pathway that may be accessible to the public and considered all alternative conceptual models for radionuclide transport analysis.
  - Resolution: (1) The pathways into which these releases occur leave the site boundary before entering the Savannah River; The NRC staff completed an independent analysis of the different groundwater pathways and confirmed that releases to the accessible environment met the requirement of 10 CFR Part 20, Appendix B - **Closed**.
  - COL Action Item 2.4-1: No chelating agents will be comingled with radioactive waste liquids and that such agents will not be used to mitigate an accidental release, or do the transport analysis with chelating agents.

Section 2.5: Geology, Seismology and Geotechnical Engineering

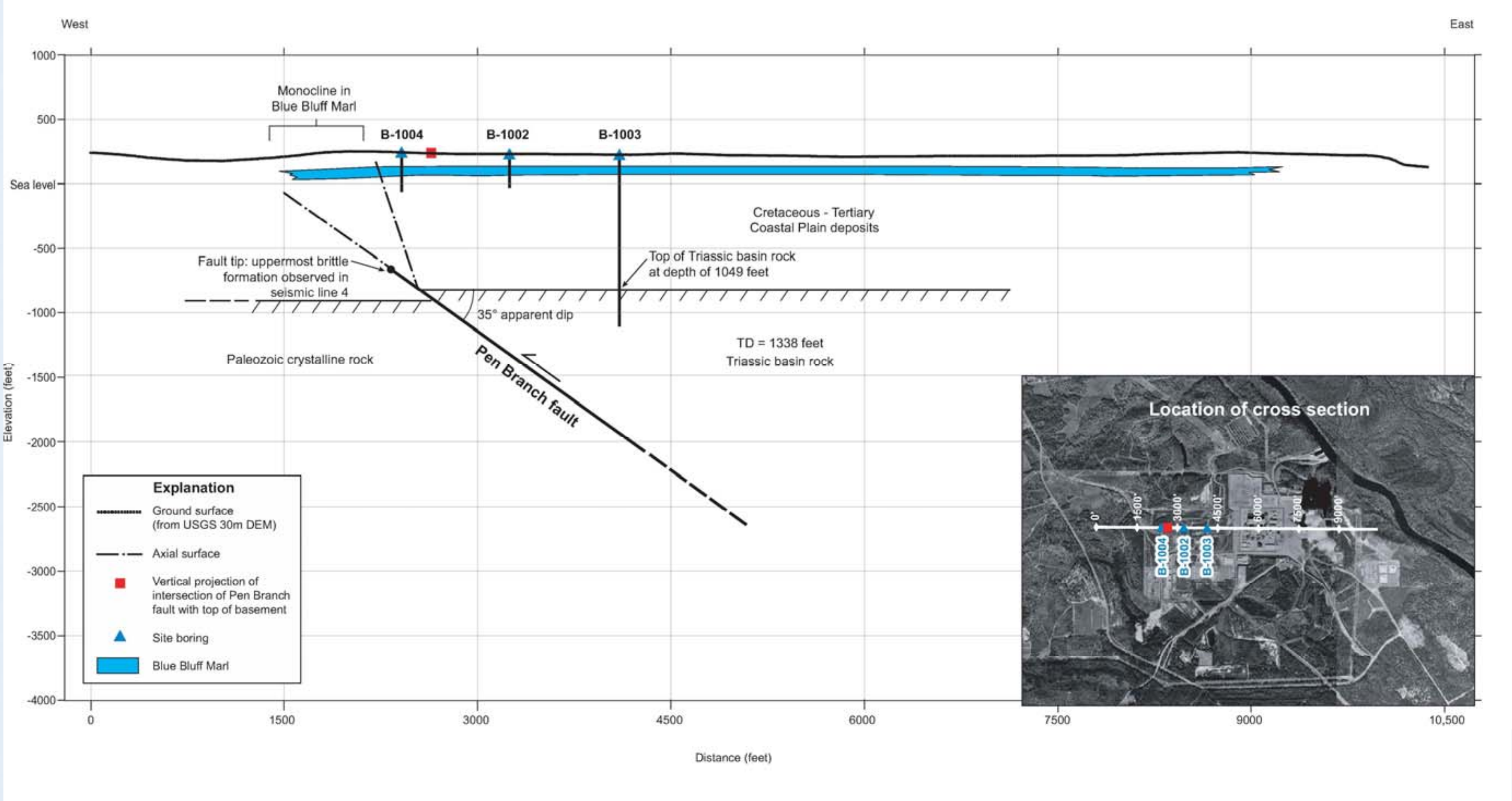
- Section 2.5.1 Site and Regional Geology
- Section 2.5.2 Vibratory Ground Motion
- Section 2.5.3 Surface Faulting
- Section 2.5.4 Stability of Subsurface Materials
- Section 2.5.5 Slope Stability

2.5.1 Basic Geologic & Seismic Information



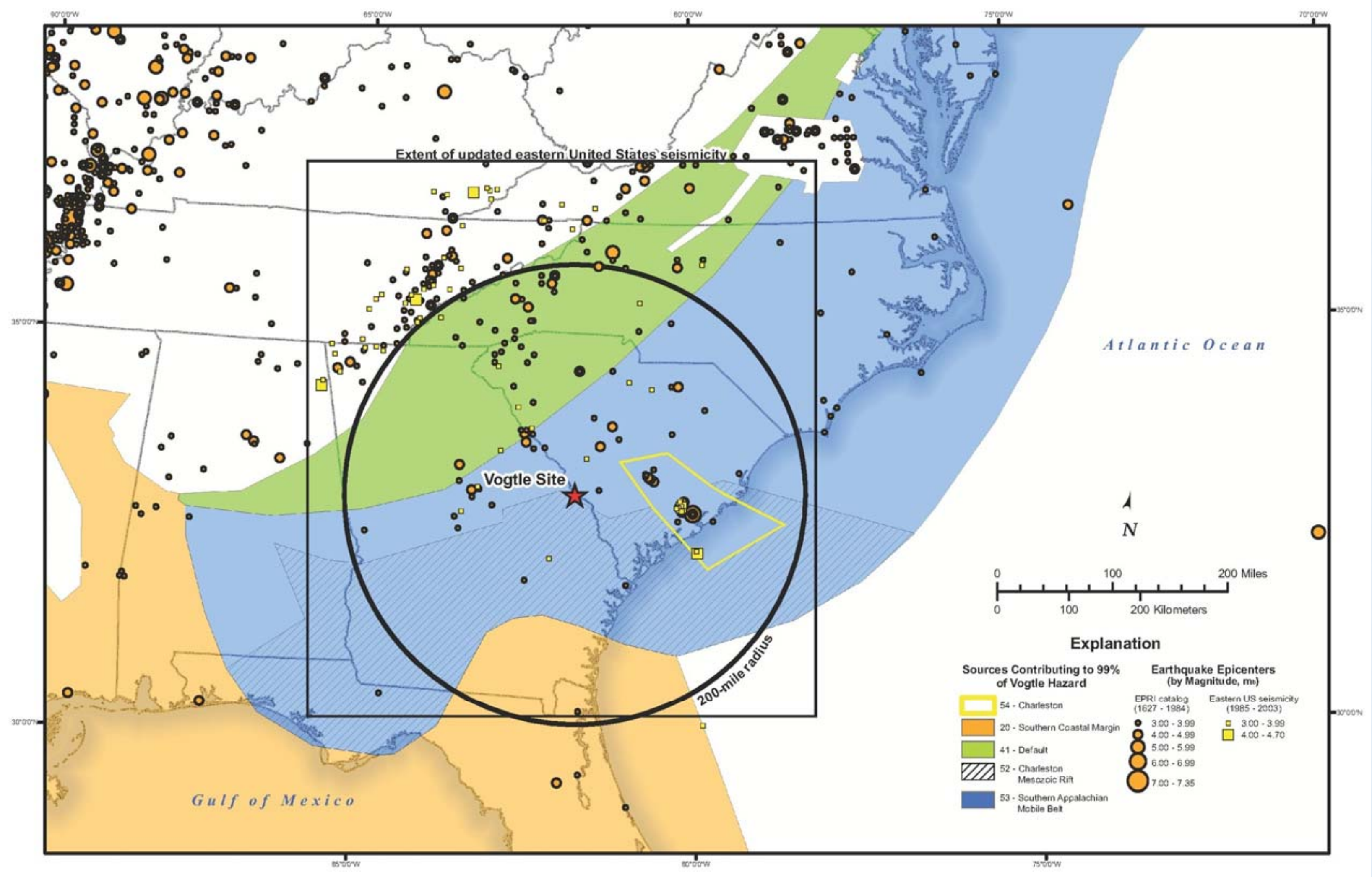
Geology in the ESP Site Vicinity

2.5.1 Basic Geologic & Seismic Information



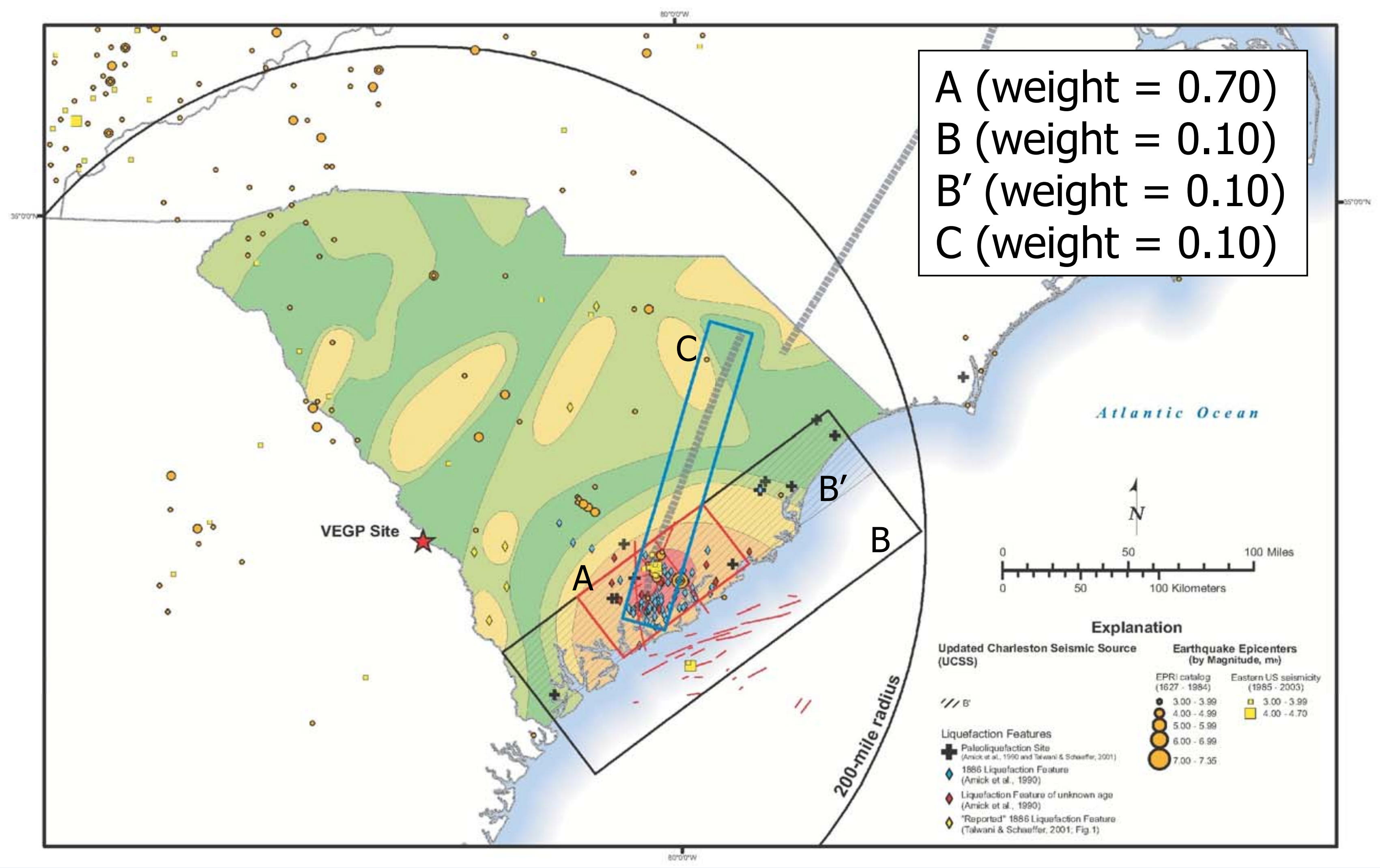
E-W Cross Section: Pen Branch Fault beneath VEGP site

2.5.2 – Vibratory Ground Motion



Example of EPRI Team Source Zones

2.5.2 Vibratory Ground Motion

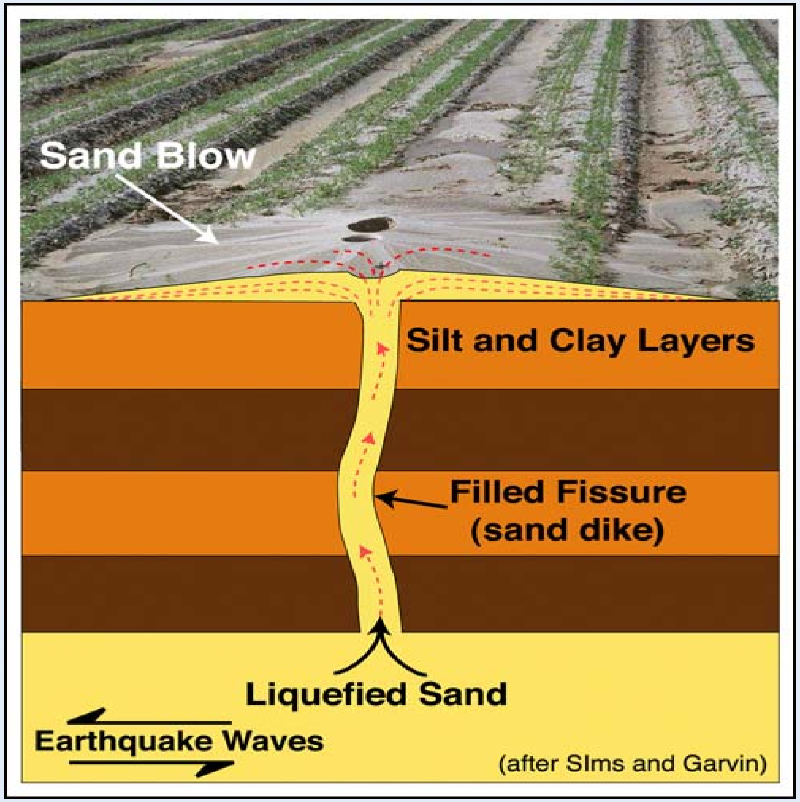


A (weight = 0.70)
B (weight = 0.10)
B′ (weight = 0.10)
C (weight = 0.10)

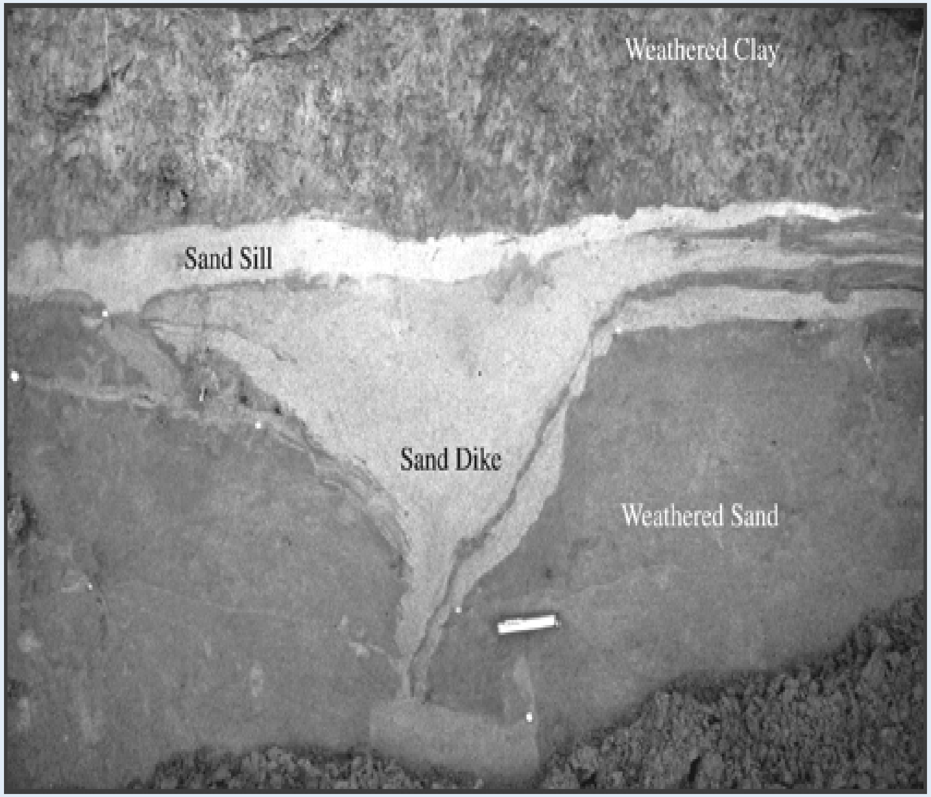
Updated Charleston Seismic Source

Charleston Update

- Charleston update based on liquefaction features from historic and prehistoric earthquakes
- Liquefaction features occur in response to strong ground shaking



(after Sims and Garvin)



Geology and Seismology

- **3 Significant Open Items addressing:**
  - **Dames and Moore EPRI-SOG Team source model**
  - **Eastern Tennessee Seismic Source Zone model**
  - **Presence of Injected Sand Dikes in site area**

2.5.4 Stability of Subsurface Material and Foundations

- **Engineering Properties of Soils and Rocks**
- **Site Explorations**
- **Geophysical Surveys**
- **Liquefaction Potential**
- **Static Stability**

2.5.4 Stability of Subsurface Material and Foundations

- **12 Open Items addressing the adequacy of:**
  - **Field and Laboratory Testing of Subsurface Materials**
  - **Measurements of Shear Wave Velocity**
  - **Development of Soil Degradation and Damping Ratio Curves**
- **Permit Condition added to require removal of Upper Sand Layer**
- **12 COL Action Items -** Resolved

2.5.4 Stability of Subsurface Material and Foundations

Site Investigations	ESP	LWA
Borings	14	174
CPTs	10	21
Test Pits	0	8
Observation Wells	15	0
P-S Velocity Logs	5	6

SER Section 13.3: Emergency Planning

- First complete EP review under 10 CFR Part 52
- Complete & Integrated Emergency Plan (ESP)
  - Included FEMA review of State/local plans
- First-of-a-kind EP Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) (30 ITAs/106 ACs)
- SER with Open Items (13 EP Open Items, 3 COL Action Items)
- Advanced SER (no EP Open Items, no EP COL Action Items, 7 EP Permit Conditions)

SER Section 13.3: Emergency Planning

SER Open Item 13.3-4 (EALs)

- NEI 07-01 EALs (AP1000 & ESBWR) (ongoing NRC endorsement review of NEI 07-01)
- AP1000 DCD EALs apply to Units 3 & 4
- Related Westinghouse amendments to AP1000 DCD (ongoing NRC AP1000 DCD review under docket 52-006)
- EAL resolution via 6 Permit Conditions (2 through 7)

SER Section 13.3: Emergency Planning

Permit Conditions:

- Emergency Action Levels (EALs)
  - 2 & 3 – NEI 07-01
  - 4 & 5 – AP1000 DCD Amendments (Units 3 & 4 TSC)
  - 6 & 7 – Full EAL set based on as-built plant, State/local agreed, & NRC approved (10 CFR Part 50, App. E.IV.B)
  - ITAAC 1.1.2 – EAL scheme consistent with RG 1.101
    - RG 1.101 is expected to endorse NEI-07-01
- Technical Support Center (TSC)
  - 8 – TSC location (AP1000 DCD, Tier 2* amendment)

SER Section 13.3: Emergency Planning

Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC):

- Planning Standard (10 CFR 50.47(b)(4))
  - A standard emergency classification & action level scheme, the bases of which include facility system and effluent parameters, . . .
- EP Program Element (NUREG-0654, evaluation criterion D.1)
  - An emergency classification & EAL scheme must be established . . . The specific instruments, parameters or equipment status shall be shown for establishing each emergency class, in the in-plant emergency procedures. The plan shall identify the parameter values and equipment status for each emergency class.
- Inspections, Tests, Analysis (ITA)
  - 1.1.2 – An analysis of the EAL technical bases will be performed to verify as-built, site-specific implementation of the EAL scheme.
- Acceptance Criteria (AC)
  - 1.1.2 – The EAL scheme is consistent with Regulatory Guide 1.101 [which is expected to endorse NEI 07-01 following staff review, including AP1000-related ITAAC]

UNITED STATES NUCLEAR REGULATORY COMMISSION

Protecting People and the Environment

Presentation to the ACRS Full Committee

Safety Review of the Vogtle Electric Generating Plant Limited Work Authorization Request

December 4, 2008

Vogtle LWA Request

Requested Activities:

- Placement of engineered backfill
- Retaining walls
- Lean concrete backfill
- Mudmats
- Waterproof membrane

2.5.4 Stability of Subsurface Materials and Foundations

LWA Key Issues

- Adequacy of borings at the site
- Geotechnical engineering properties of the subsurface materials, especially the Blue Bluff Marl and Lower Sand Stratum
- Backfill Specifications

2.5.4 Stability of Subsurface Materials and Foundation Interfaces

LWA Key Issues – Backfill ITAAC

Design Requirement	Inspections and Tests	Acceptance Criteria
Backfill material under Seismic Category 1 structures is installed to meet a minimum of 95 percent modified Proctor compaction.	Required testing will be performed during placement of the backfill materials.	A report exists that documents that the backfill material under Seismic Category 1 structures meets the minimum 95 percent modified Proctor compaction.
Backfill shear wave velocity is greater than or equal to 1,000 fps at the depth of the nuclear island foundation and below.	Field shear wave velocity measurements will be performed when backfill placement is at the elevation of the bottom of the Nuclear Island foundation and at finish grade.	A report exists and documents that the as-built backfill shear wave velocity at the nuclear island foundation depth and below is greater than or equal to 1,000 fps.

2.5.4 Stability of Subsurface Materials and Foundations

Section 2.5.4 Conclusions

- Adequacy of borings
  - Performed substantially more borings
- Geotechnical Engineering properties of subsurface materials
  - Significant additional site investigations provided sufficiently detailed information
- Backfill Specifications
  - Test Pad measurements of backfill properties
  - ITAAC to verify compaction density and shear wave velocity

Scope of Review for Chapter 3

SRP 3.7.1-Seismic Design Parameters

- Vibratory Ground Motion
- Critical Damping
- Supporting Media (pertaining to SSI modeling)

SRP 3.7.2- Seismic Systems Analysis

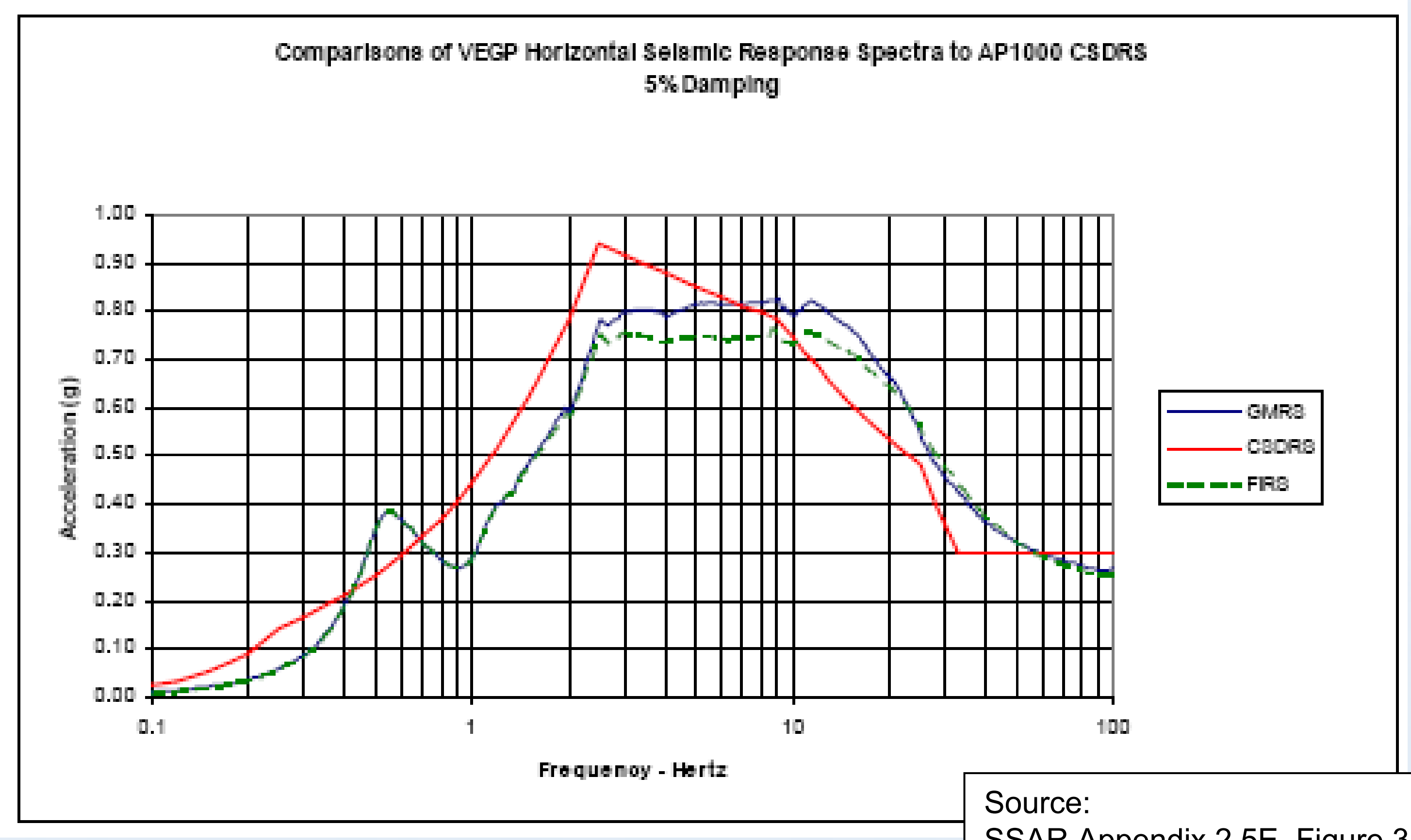
- Seismic Model Description
- Soil-Structure-Interaction Analysis

SRP 3.8.5-Foundations

- Foundation Stability
  - Sliding
  - Overturning

SER Section 3.7.1
Seismic Design Parameters

Comparison of Vogtle Horizontal GMRS and FIRS with AP1000 CSDRS



Source:
SSAR Appendix 2.5E, Figure 3-4

SER Section 3.7.1 Seismic Design Parameters

Technical Evaluation/Findings

Vibratory Ground Motion

- Approximate method was used for developing the FIRS. Review indicates that the method results in a conservative estimate of horizontal seismic demand.

- The FIRS defined as an outcrop motion in the free field satisfied the minimum PGA value of 0.10g (10 CFR Part 50, Appendix S)

Critical Damping

- The critical structural damping values used in SSI analysis were consistent with damping values provided in RG 1.61.

Supporting Media

- SSI modeling assumptions properly account for site characteristics such as depth of soil over bedrock, soil properties, soil layering characteristics and groundwater elevation.

SER Section 3.7.2
Seismic Systems Analysis

Technical Evaluation/Findings

Seismic Model

- The use of 2D SASSI models is acceptable for the evaluation of sliding stability and bearing pressure demands.

Soil-Structure-Interaction Analysis

- Staff compared the analysis results (e.g., ZPA values near the NI center-of-gravity) with the AP1000 DCD soft soil case and found them to be similar.

- Maximum seismic base shear forces are acceptable based on staff simplified independent calculations.

SER Section 3.8.5 Foundations

Summary of Application

- Test data of waterproofing membrane indicate a coefficient of friction of 0.7 between the membrane and the concrete mudmat.
- Test data indicate a coefficient of friction of 0.45 for soil immediately below mudmat.
- Soil test data indicate a bearing capacity of 42 ksf.

SER Section 3.8.5 Foundations

Technical Evaluation/Findings

NI Structure Stability Analysis

- Staff reviewed the maximum horizontal seismic forces and maximum friction forces below the basemat.

Maximum NI Seismic Forces

Reaction	Vogtle Lower Bound	Vogtle Best Estimate	Vogtle Upper Bound
Seismic Shear NS	78.3 E3 kips	82.5 E3 kips	89.0 E3 kips
Seismic Shear EW	88.9 E3 kips	89.8 E3 kips	95.8 E3 kips
Friction Force	117.3 E3 kips	116.7 E3 kips	116.4 E3 kips

- The NI structure will not slide during the SSE, because the frictional force is greater than the inertial force.

SER Section 3.8.5 Foundations

Technical Evaluation/Findings (Continued)

Bearing Capacity

- The maximum dynamic bearing pressure on soils for the NI, radwaste, annex, and turbine buildings are 17.95 ksf, 1.68 ksf, 7.20 ksf, and 2.54 ksf, respectively, during the SSE.

- The minimum factor of safety with respect to a failure of the dynamic soil bearing capacity during the SSE is 2.34 (42 ksf divided by 17.95).

Summary Findings

SRP Section 3.7.1 Seismic Design Parameters

- Adequately developed seismic design parameters.
- Met the applicable regulatory requirements.

SRP Section 3.7.2 Seismic Systems Analysis

- Adequately performed site-specific 2D SSI analysis for the purpose of determining the maximum seismic demands for use in the NI structure stability and maximum dynamic soil bearing evaluations.
- Staff's evaluation of in-structure response will be done as part of the SCOL review.
- Met the applicable regulatory requirements.

SRP Section 3.8.5 Foundations

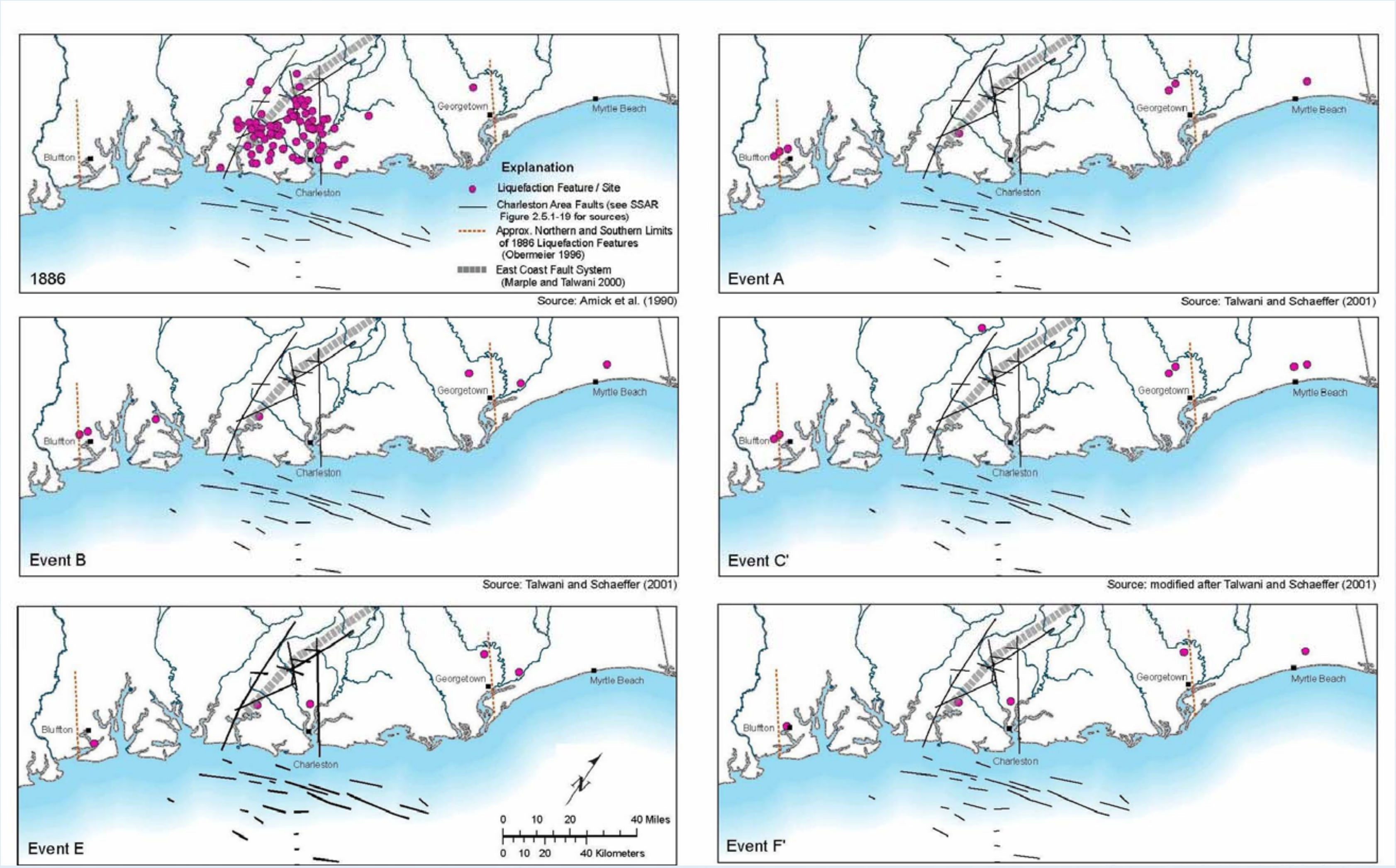
- Demonstrated that the mudmat and the waterproofing membrane are adequate and that the NI foundation is stable during an SSE.
- Met the applicable regulatory requirements.

Advanced SER/LWA Conclusions

- The VEGP ESP application meets the applicable standards and requirements of the Act and the Commission’s regulations.
- Site Characteristics, Design Parameters, and Terms and Conditions proposed to be included in the Permit meet the applicable requirements of Part 52.
- There is reasonable assurance that the site is in conformity with the provisions of the Act, and the Commission’s regulations.
- The proposed ITAAC are necessary and sufficient, within the scope of the ESP, to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the emergency plans, the provisions of the Act, and the Commission’s regulations.
- Issuance of the permit will not be inimical to the common defense and security or to the health and safety of the public

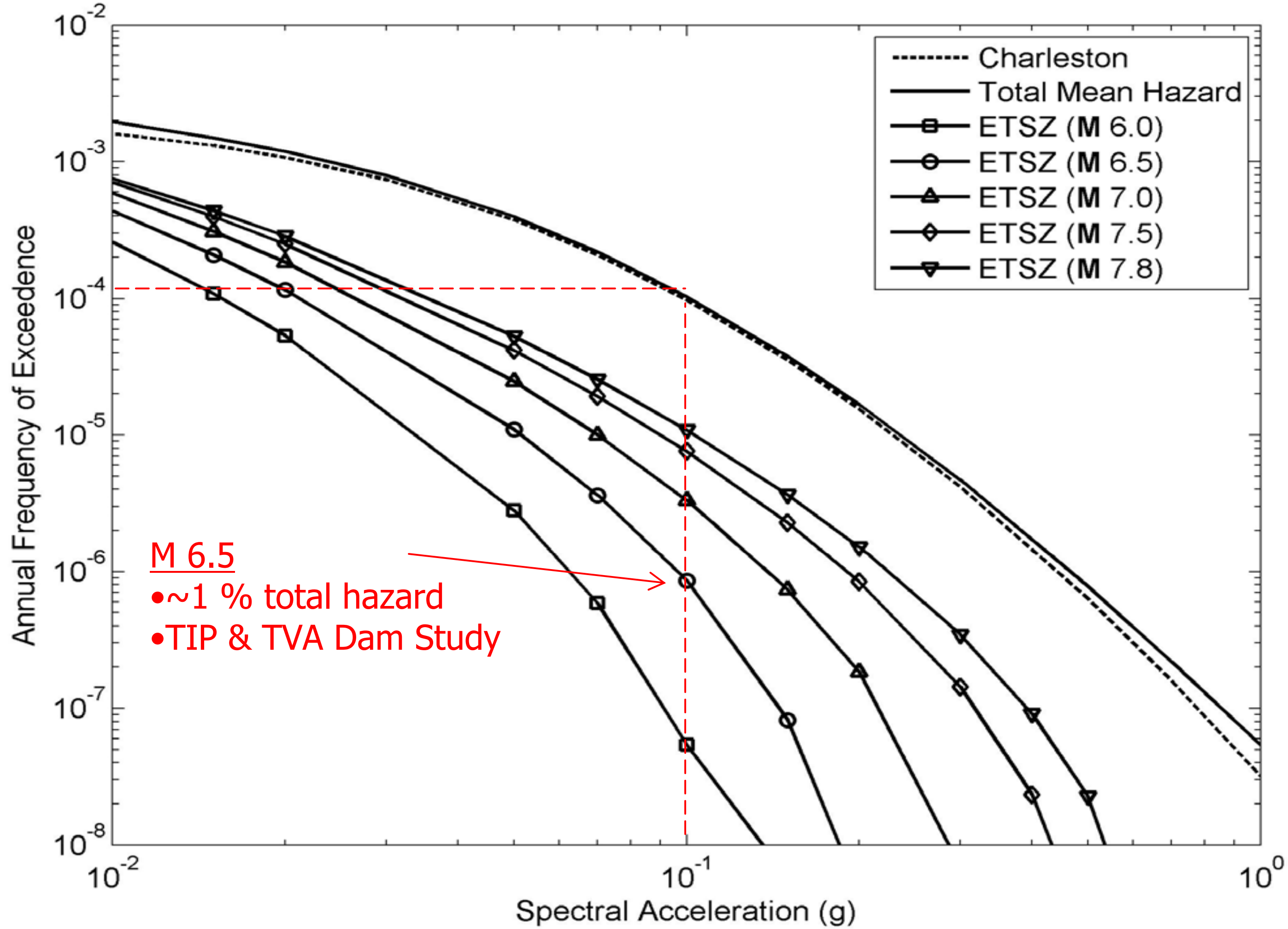
BACKUP SLIDES for ESP

2.5.2 Vibratory Ground Motion



Distribution of Charleston Source Paleoliquefaction Features

2.5.2 Vibratory Ground Motion



M 6.5

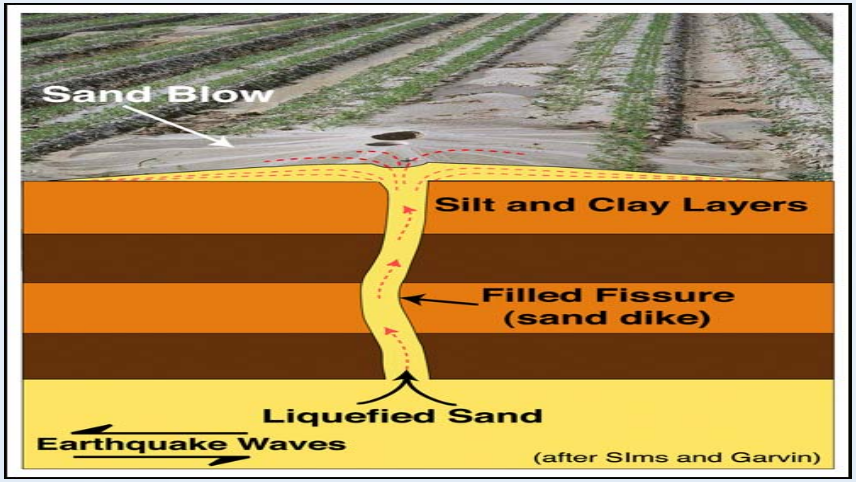
- ~1 % total hazard
- TIP & TVA Dam Study

Results of Staff's ETSZ Sensitivity Study

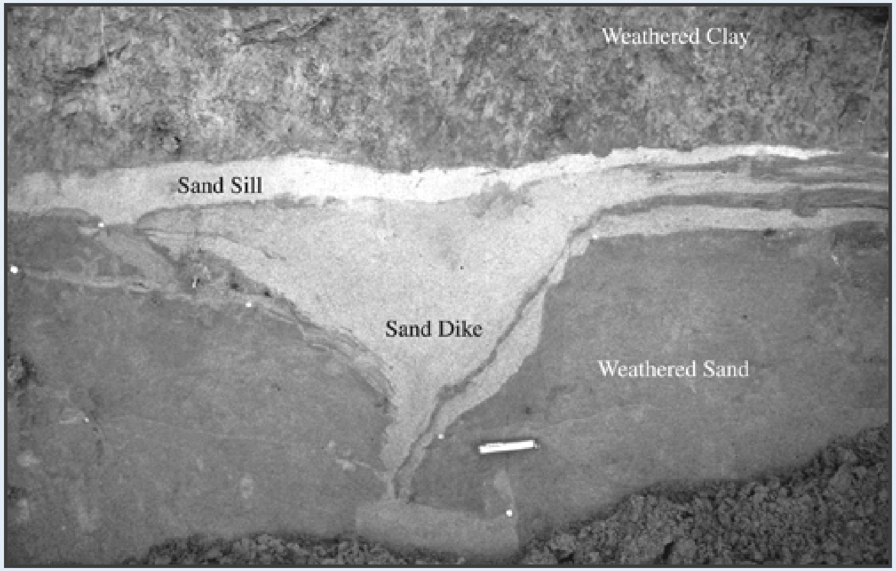
2.5.2 Vibratory Ground Motion

- **Liquefaction**

  - Liquefaction features occur in response to strong ground shaking
  - Liquefaction susceptibility is a function of site characteristics
  - Liquefaction features commonly occur in the form of sand blows

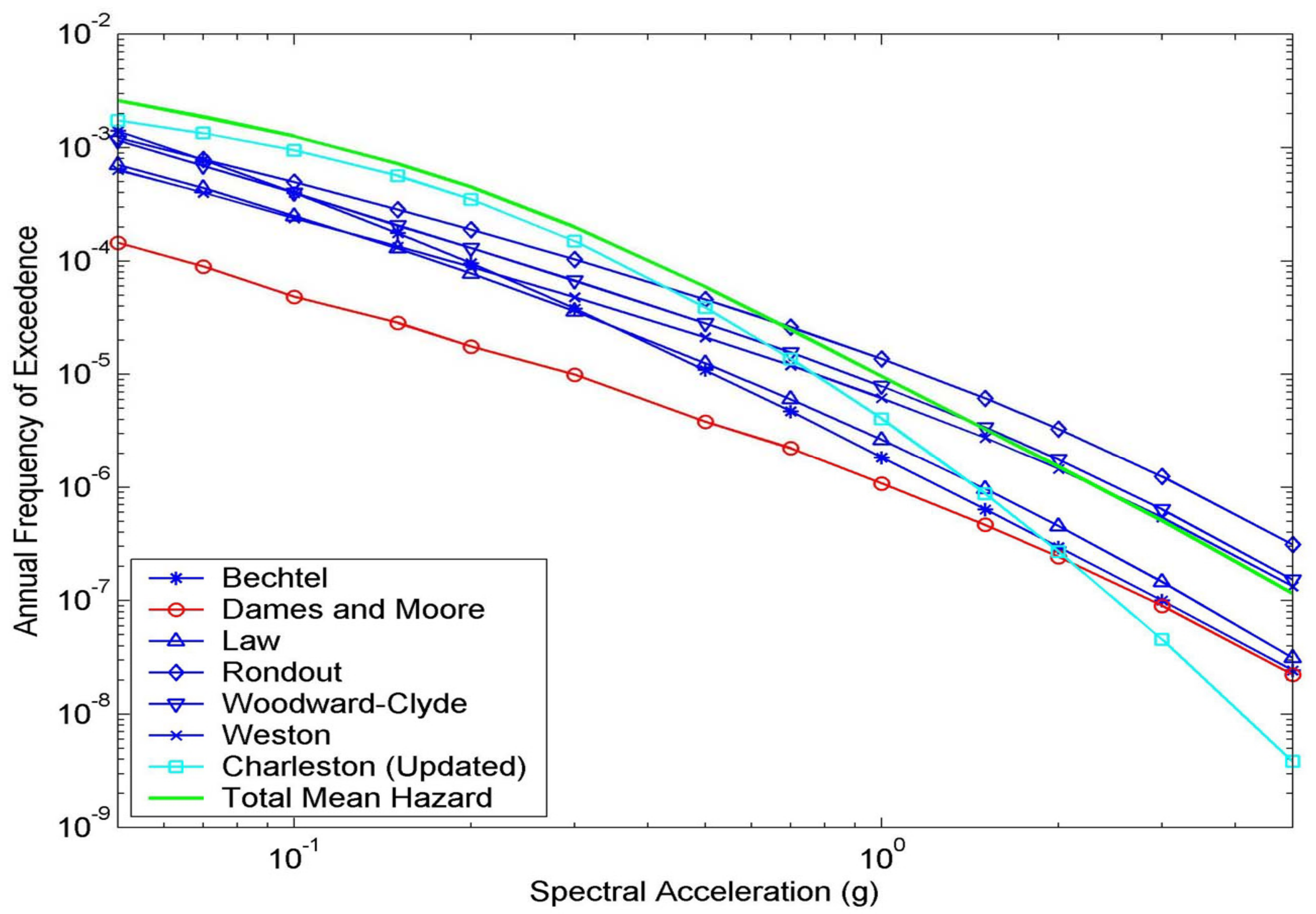


(after Sims and Garvin)



2.5.2 Vibratory Ground Motion

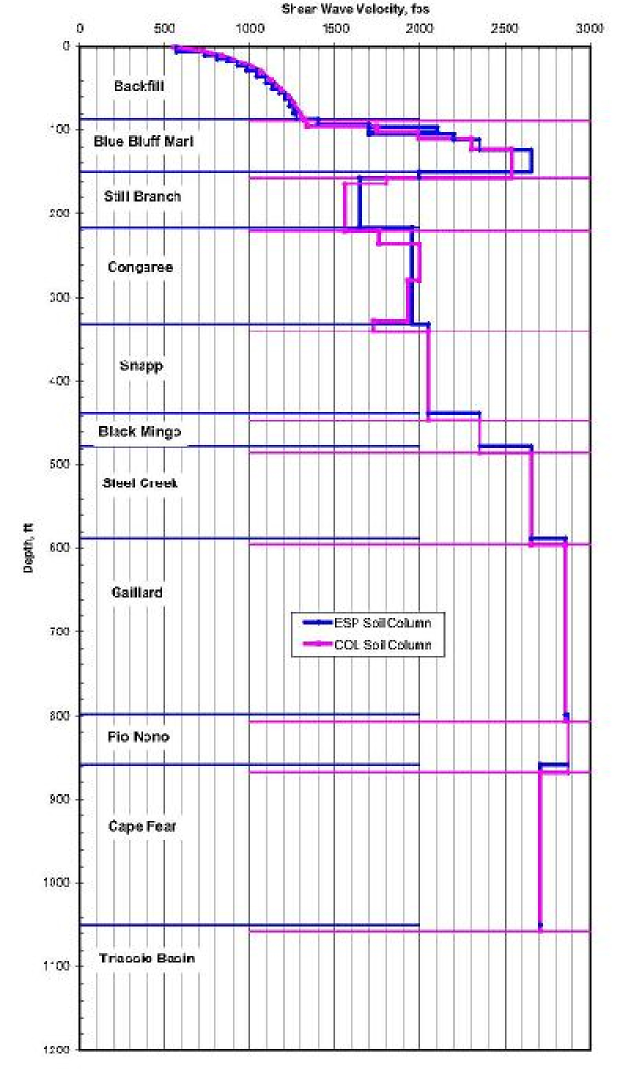
10-Hz Total Mean Hazard Curve



2.5.4 Stability of Subsurface Material and Foundations

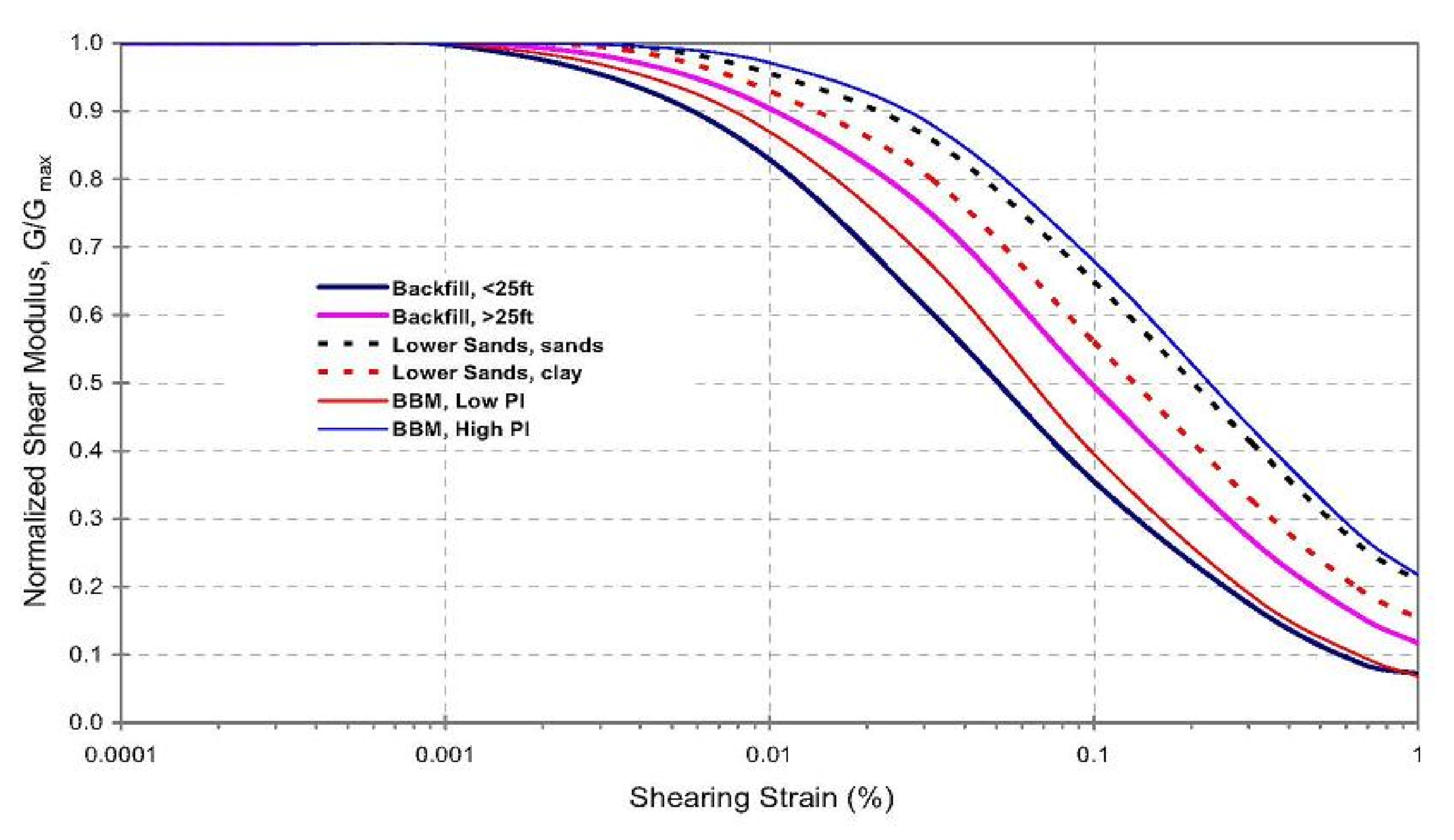
Shear Wave

Velocity Profile



2.5.4 Stability of Subsurface Material and Foundations

Shear Modulus Reduction Curve

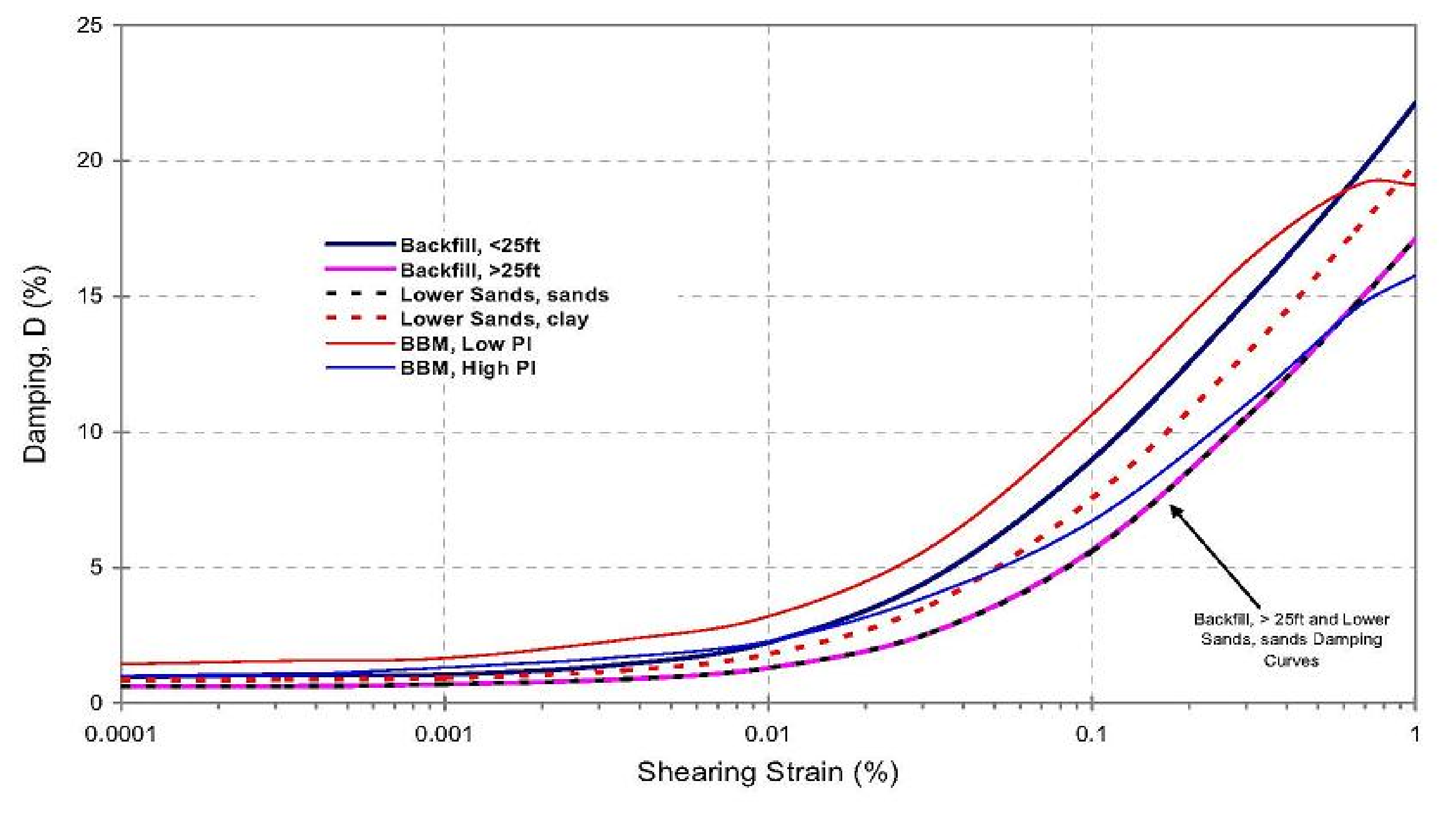


December 4, 2008

UNITED STATES NUCLEAR REGULATORY COMMISSION

2.5.4 Stability of Subsurface Material and Foundations

Damping Ratio Curves



Backfill, > 25ft and Lower Sands, sands Damping Curves

2.5.2 Vibratory Ground Motion

- **Charleston Liquefaction Features**
  - Abundant liquefaction features from historic and prehistoric earthquakes were mapped for ~130mi. NE-SW along the South Carolina coast and >65mi. inland from coast
  - Paleoliquefaction features formed during prehistoric earthquakes

UNITED STATES NUCLEAR REGULATORY COMMISSION

2.5.2 Vibratory Ground Motion

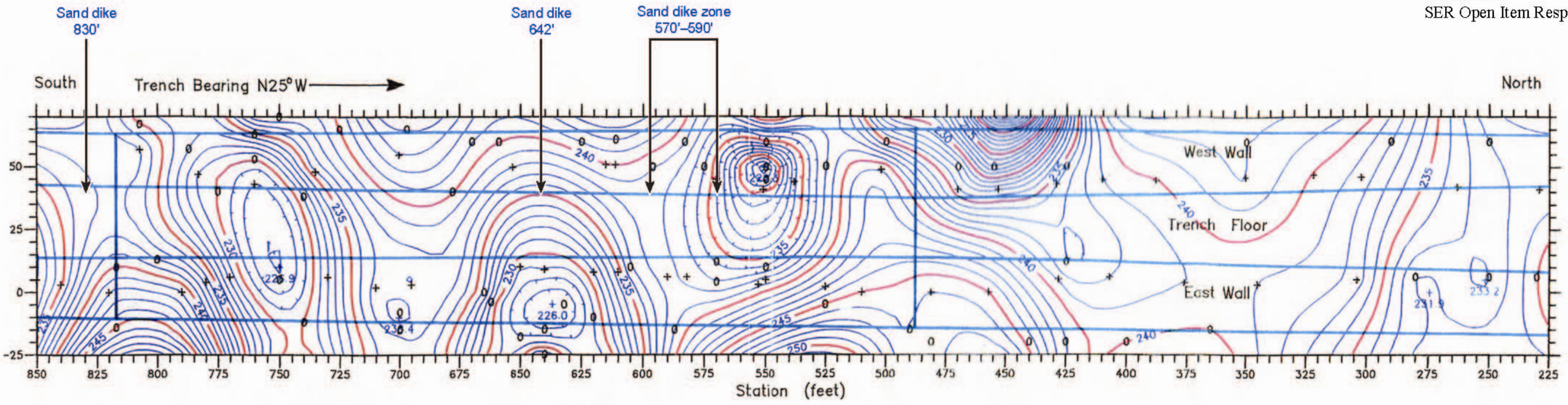
Illustrations of historic 1886 liquefaction features from the Charleston Area

2.5.2 Vibratory Ground Motion

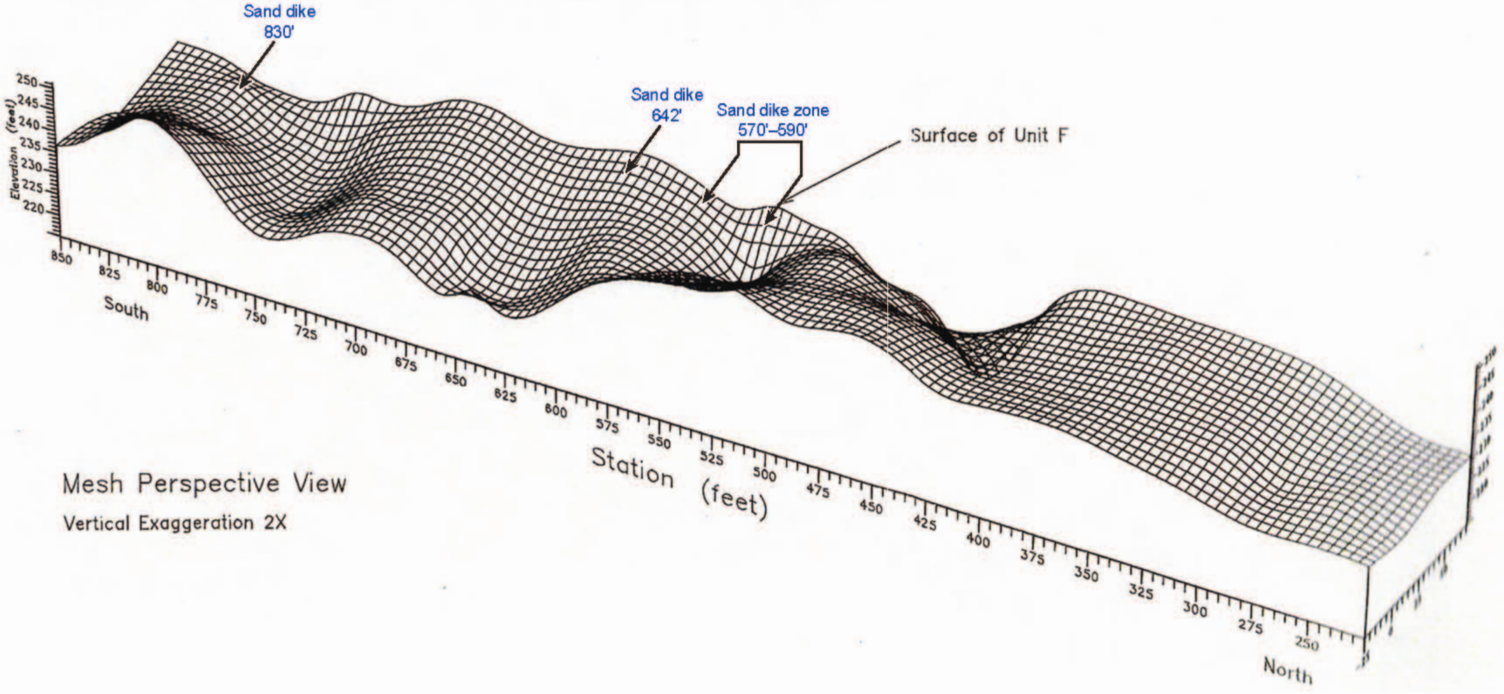
- **Charleston Paleoliquefaction Features**
  - Paleoliquefaction features, documented since the 1989 EPRI study, contributed to the update of the Charleston source zone
  - Liquefaction features represent 5 similar magnitude earthquakes (in addition to 1886) during the past ~5000 years
  - Estimated repeat times for large earthquakes in the Charleston area:
    - 500-600 years, based on a complete 2,000 yr history
    - 900-1000 yrs, based on a complete 5,000 yr history

BACKUP SLIDES

Enclosure 1
SER Open Item Respons



Structure Contour Map, Surface of Unit F – Trench Plan View



Mesh Perspective View

Vertical Exaggeration 2X

Explanation

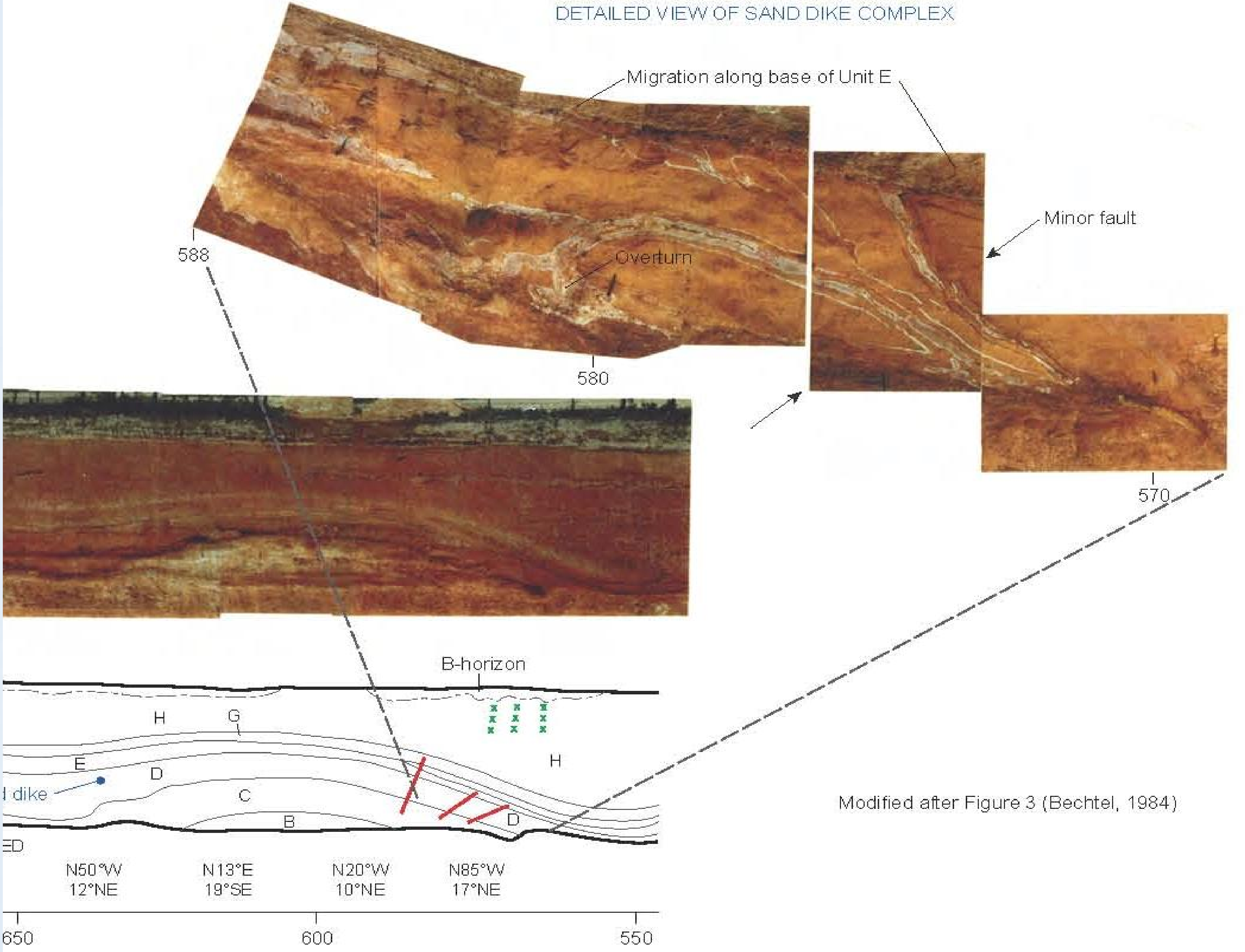
+ Surveyed Elevation

0 Elevation Extrapolated From Bedding Attitude

Contour Interval 1 Foot

Modified after Figure 5 (Bechtel, 1984)

BACKUP SLIDES



DETAILED VIEW OF SAND DIKE COMPLEX

Modified after Figure 3 (Bechtel, 1984)

2.5.4 Stability of Subsurface Materials and Foundations

- **Insufficient Laboratory Testing (Open Item 2.5-11)**
  - Issue: Conduct sufficient field & laboratory tests to reliably determine subsurface soil static & dynamic properties at the ESP site
  - Resolution: In support of the LWA request, the applicant performed additional field and laboratory investigations which were used to determine the static and dynamic properties of the subsurface materials

2.5.4 Stability of Subsurface Materials and Foundations

- **Blue Bluff Marl Load-bearing Properties (Open Item 2.5-12)**
  - Issue: Provide sufficient data to derive reliable site-specific engineering parameters for the Blue Bluff Marl
  - Resolution: The applicant performed SPT and split-spoon sampling in almost all ESP borings and conducted additional laboratory tests such as grain size distribution, Atterberg Limits, and carbonate content

2.5.4 Stability of Subsurface Materials and Foundations

- **Undrained Shear Strength (Open Item 2.5-13)**
  - Issue: Provide sufficient sampling and testing results to reliably derive the undrained shear strength and other related engineering parameters
  - Resolution: The applicant revised the SSAR using the additional field and laboratory investigations to provide the preconsolidation pressure calculations and overconsolidation ratios

2.5.4 Stability of Subsurface Materials and Foundations

- **Angles of Friction (Open Item 2.5-14)**
  - Issue: Provide reliable effective angles of internal friction for the subsurface soils
  - Resolution: The applicant revised the SSAR to include a description of the empirical correlation of average effective angles of internal friction which were used.

2.5.4 Stability of Subsurface Materials and Foundations

- **Blue Bluff Marl Behavior (Open Item 2.5-15)**
  - Issue: Provide information to demonstrate that the Blue Bluff Marl will behave as a hard clay or soft rock material
  - Resolution: Additional borings in support of the LWA request were used to demonstrate the behavior of the BBM

2.5.4 Stability of Subsurface Materials and Foundations

- **Elastic Modulus (Open Item 2.5-16)**
  - Issue: Provide sufficient site-specific data to justify the determination of the design parameter elastic modulus “E” for the Upper and Lower Sand Strata
  - Resolution: The applicant used representative data from the SPTs performed in support of the LWA request to determine E

2.5.4 Stability of Subsurface Materials and Foundations

- **Unit Weight Values (Open Item 2.5-17)**
  - Issue: Develop sufficient data (vs. values from previous investigations) to calculate the unit weight values for the ESP subsurface soils
  - Resolution: Additional data were included in support of the LWA request and were used to calculate the unit weight of the subsurface materials

2.5.4 Stability of Subsurface Materials and Foundations

- **SSAR Degradation Curve Revision (Open Item 2.5-20)**
  - Issue: Revise SSAR Sections 2.5.2.5.1.5, 2.5.4.7.2.1, and 2.5.4.7.2.2, along with associated tables and figures, to show the degradation curves only at a ≤1% cyclic shear strain
  - Resolution: The SSAR was revised accordingly.

2.5.4 Stability of Subsurface Materials and Foundations

- **Liquefaction Potential of Blue Bluff Marl (Open Item 2.5-21)**
  - Issue: Provide sufficient ESP soil property data to confirm that the Blue Bluff Marl is non-liquefiable
  - Resolution: Additional borings completed in support of the LWA were used to confirm the negligible liquefaction potential of the BBM

2.5.4 Stability of Subsurface Materials and Foundations

- **Bearing Capacity (Open Item 2.5-22)**
  - Issue: Provide appropriate bearing capacity estimates
  - Resolution: Later revisions to the SSAR in support of the LWA request included the bearing capacity calculations and settlement estimates

2.5.4 Stability of Subsurface Materials and Foundations

- **Previous COL Action items**
  - 2.5-1 A COL or CP applicant will need to confirm the absence of soft materials in the load bearing layers.
  - 2.5-2 A COL or CP applicant will need to confirm the locations of the soft zones and evaluate the potential impact of the soft zones on the foundation and structures.
  - 2.5-3 A COL or CP applicant will need to provide chemical test results on the backfill.

2.5.4 Stability of Subsurface Materials and Foundations

- **Previous COL Action items**
  - 2.5-4 A COL or CP applicant will need to submit plot plans and profiles of all seismic Category I facilities for comparison with the subsurface profile and material properties.
  - 2.5-5 A COL or CP applicant will need to provide detailed excavation and backfill plans during the COL stage.
  - 2.5-6 A COL or CP applicant will need to provide sufficient information to show the backfills meet the minimum shear wave requirement.

2.5.4 Stability of Subsurface Materials and Foundations

- **Previous COL Action items**
  - 2.5-7 A COL or CP applicant will need to submit ground water condition evaluations and a detailed dewatering plan during the COL stage.
  - 2.5-8 A COL or CP applicant will need to demonstrate quantitatively whether the observed large settlement that occurred at the existing VEGP units will occur at the ESP site and have no impact on the new units.
  - 2.5-9 A COL or CP applicant will need to provide more details regarding the bearing capacity during the COL stage.

2.5.4 Stability of Subsurface Materials and Foundations

- **Previous COL Action items**
  - 2.5-10 A COL or CP applicant will need to describe the design criteria and design methods, including the factor of safety for slope stability at the COL stage.
  - 2.5-11 A COL or CP applicant will need to provide information regarding ground improvement after removal of Upper Sand Stratum for the ESP site.

Key Review Areas

Chapter 15 – Radiological Consequences of Design Basis Accidents

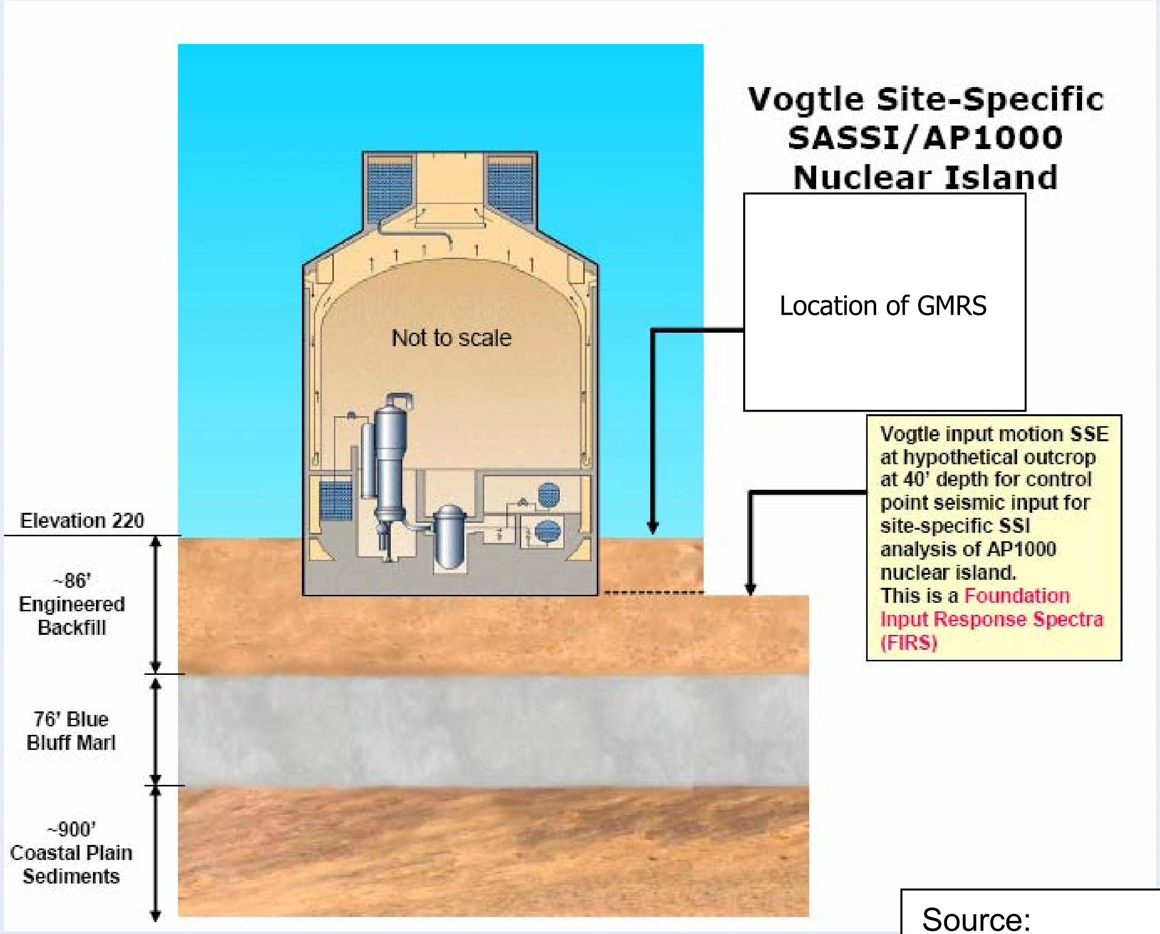
- Permit condition 9:
  - The permit will include the time-dependent isotropic release (source term) for each DBA
  - COL applicant referring to certified design only required to demonstrate site-specific atmospheric dispersion factor values less than used in DCD to show compliance with Part 100, 10 CFR 52.79 and GDC-19
  - Permit condition to not require holder of Vogtle ESP do anything more than any other COL applicant referring to a certified design. If ESP holder does not refer to a certified design, COLA would demonstrate that plant source term is bounded by the source term in ESP

Key Review Areas

- Applicant used AP1000, DCD Rev. 15
  - Calculated site-specific short term atmospheric dispersion factors (χ/Qs)
  - Ratio of site-specific to design reference χ/Qs applied to DCD calculated DBA dose to give estimate of site-specific DBA dose for each DBA in AP1000 DCD
  - Since each site-specific χ/Q was less than comparable design reference χ/Q, then site-specific DBA doses are less than AP1000 DCD DBA doses and therefore meet regulatory criteria
    - Can confirm by taking AP1000, Rev. 15 source term release rates for each DBA and calculating site-specific DBA dose using site-specific χ/Qs

Backup Slides (3.7 and 3.8)

Nuclear Island Supporting Media



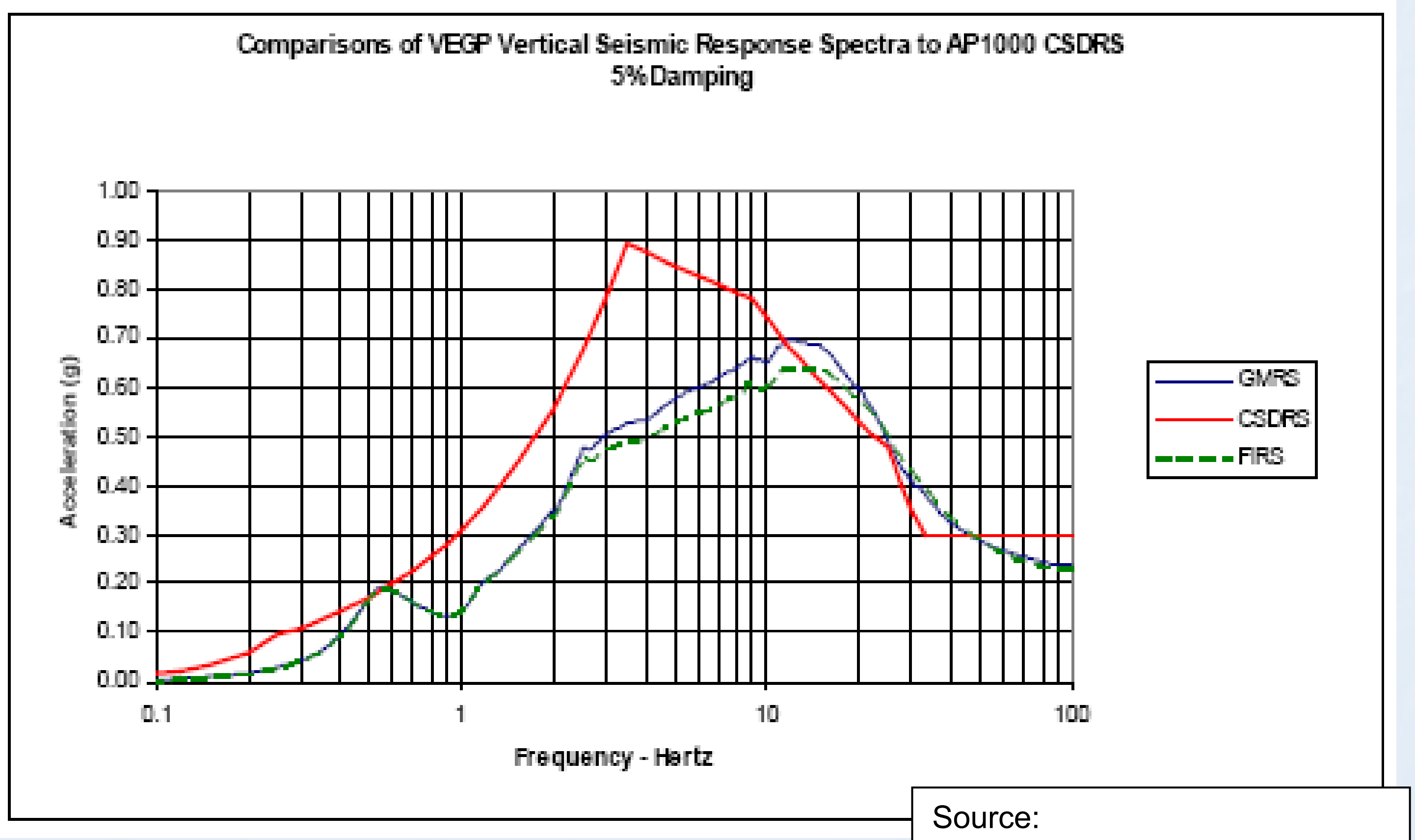
Vogtle Site-Specific SASSI/AP1000 Nuclear Island

Vogtle input motion SSE at hypothetical outcrop at 40' depth for control point seismic input for site-specific SSI analysis of AP1000 nuclear island.
This is a Foundation Input Response Spectra (FIRS)

Source:
SSAR Appendix 2.5E, Figure 5.0-2

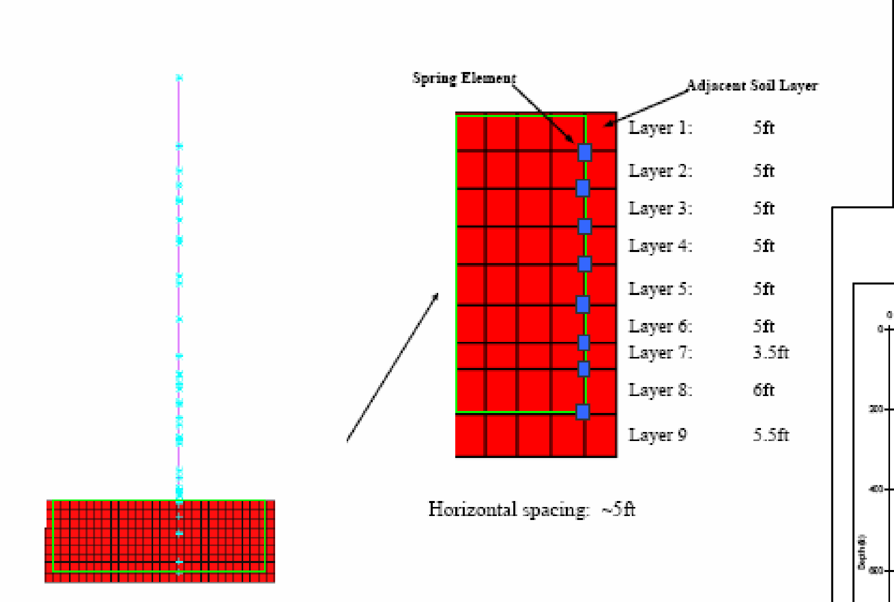
SER Section 3.7.1
Seismic Design Parameters

Comparison of Vogtle Vertical GMRS and FIRS with AP1000 CSDRS

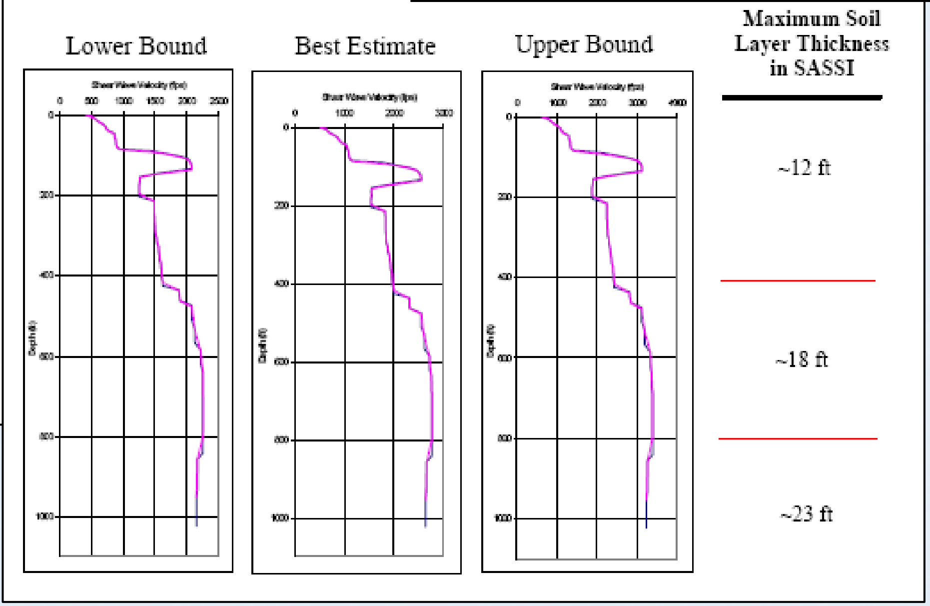


Source:
SSAR Appendix 2.5E, Figure 3-4

SSI Model

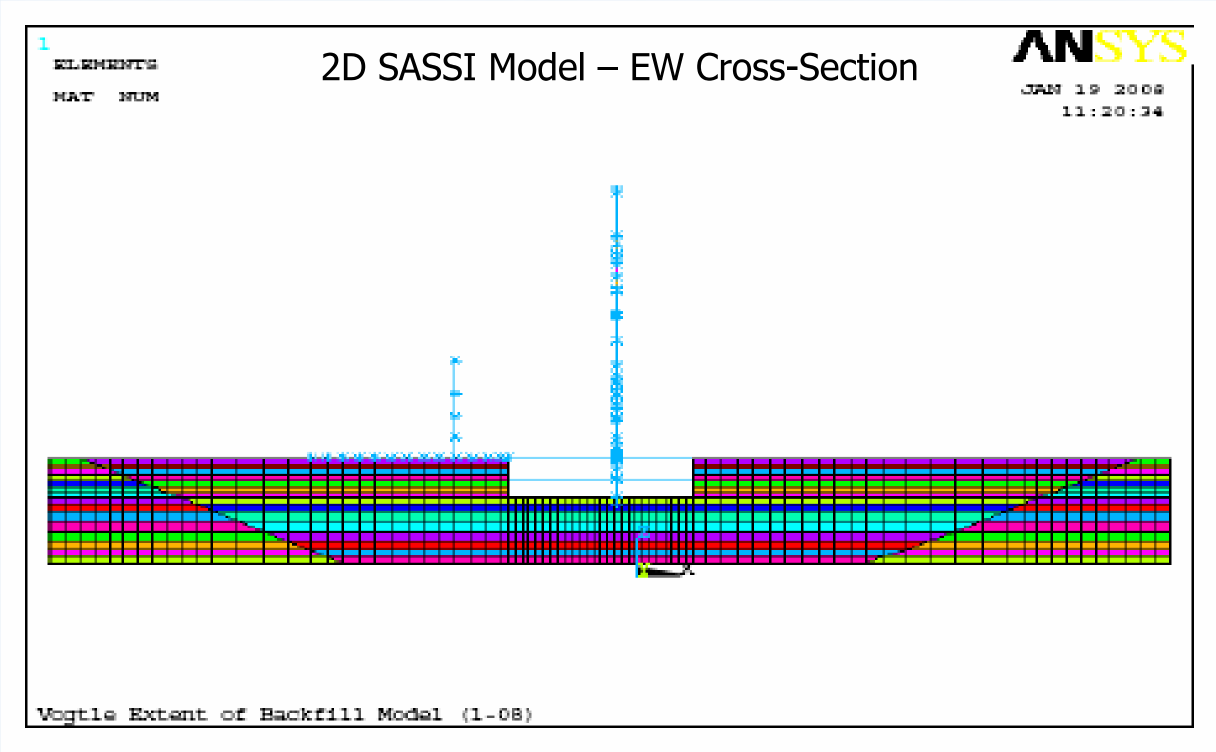


Source:
SSAR Appendix 2.5E, Figure 4.1-1



Source:
SSAR Appendix 2.5E, Figure 4.1-2

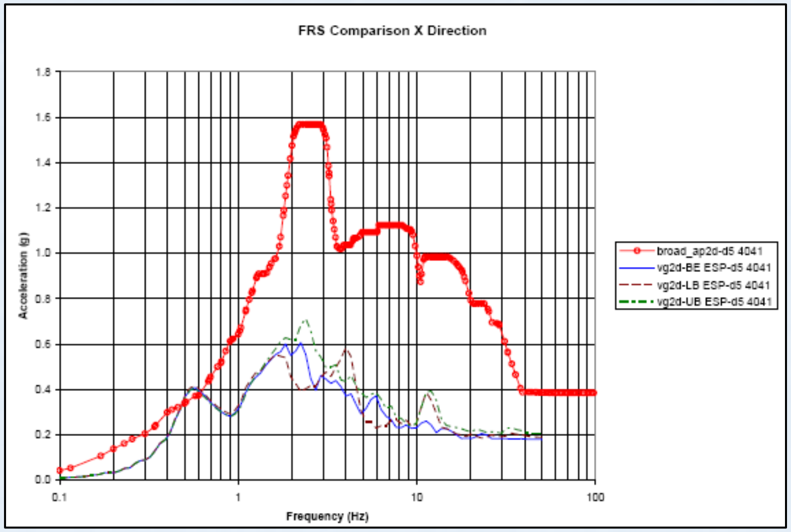
Lateral Extent of Backfill

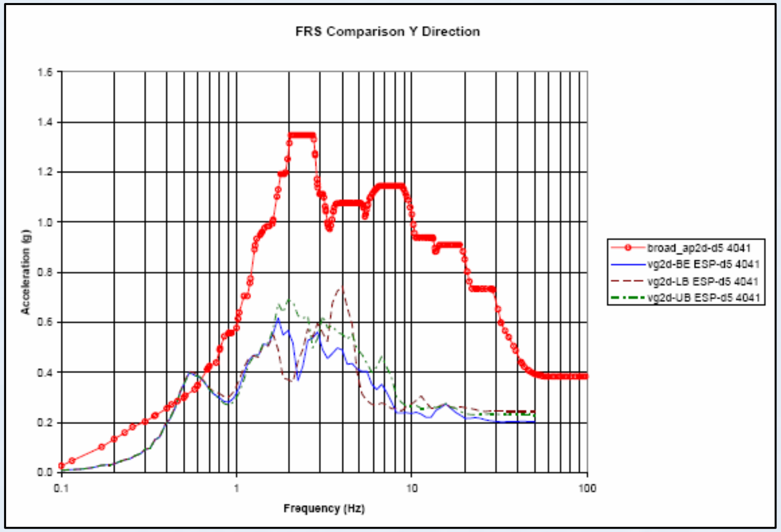

2D SASSI Model – EW Cross-Section

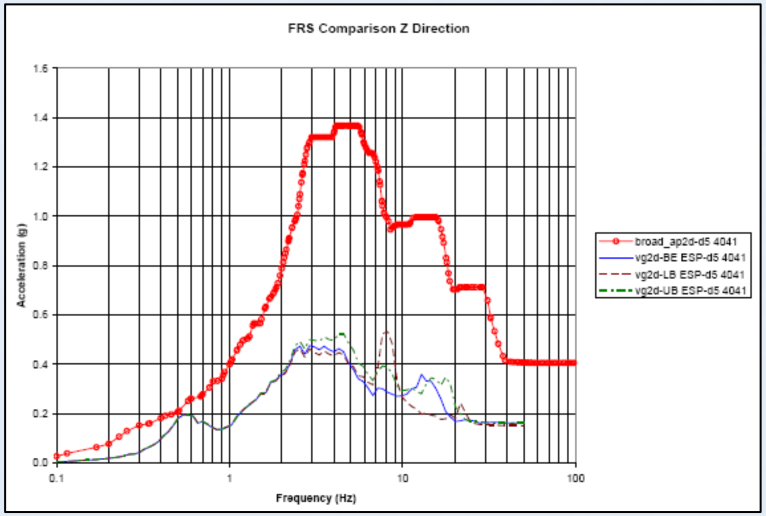
Source:
SSAR Appendix 2.5E, Figures A-1

2D Model Results

Reactor Vessel Support; EL 99 ft, Node 4041



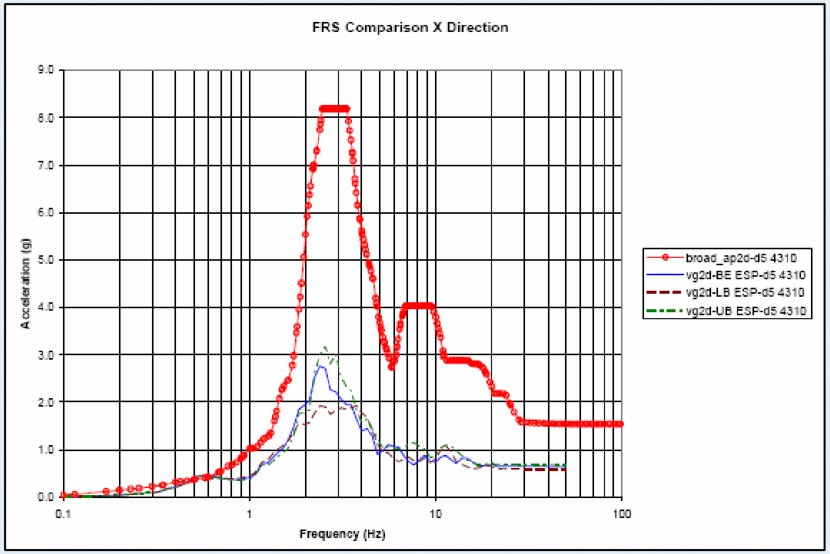


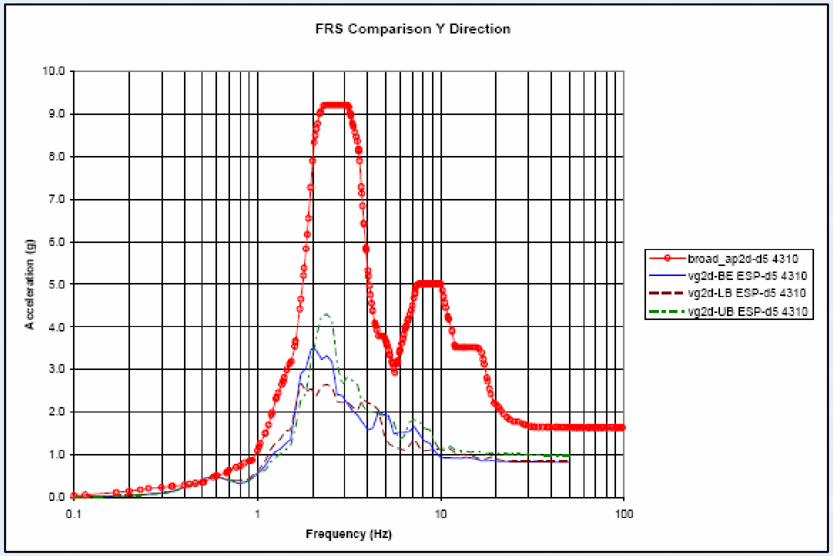


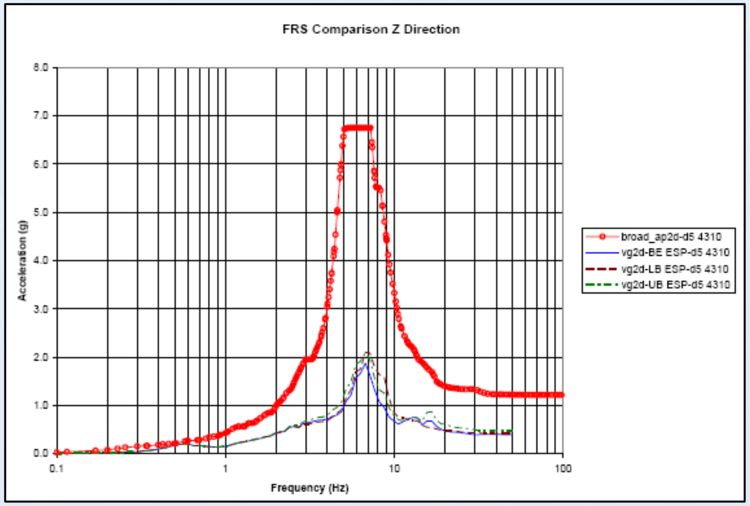
Source:
SSAR Appendix 2.5E, Figures 5.1-1,2,3

2D Model Results

Shield Building Roof; EL 327 ft, Node 4310

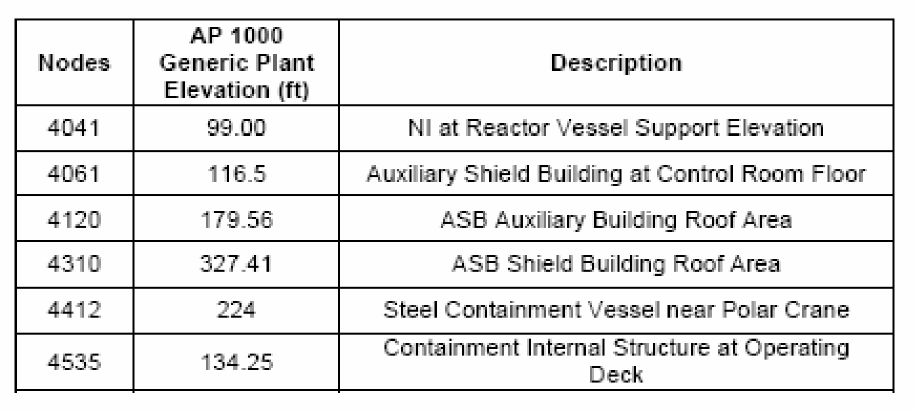






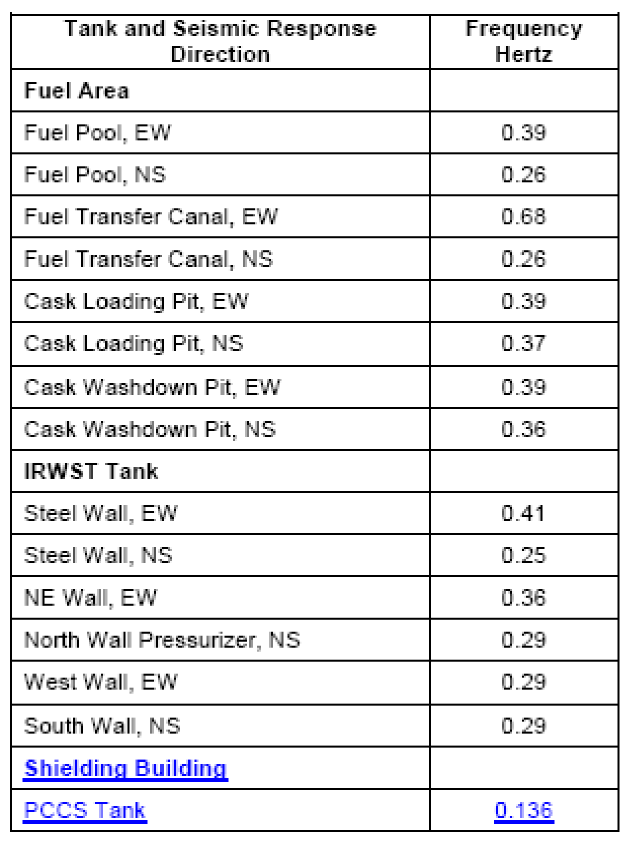
Source:
SSAR Appendix 2.5E, Figures 5.1-10,11,12

Tables



Nodes	AP 1000 Generic Plant Elevation (ft)	Description
4041	99.00	NI at Reactor Vessel Support Elevation
4061	116.5	Auxiliary Shield Building at Control Room Floor
4120	179.56	ASB Auxiliary Building Roof Area
4310	327.41	ASB Shield Building Roof Area
4412	224	Steel Containment Vessel near Polar Crane
4535	134.25	Containment Internal Structure at Operating Deck

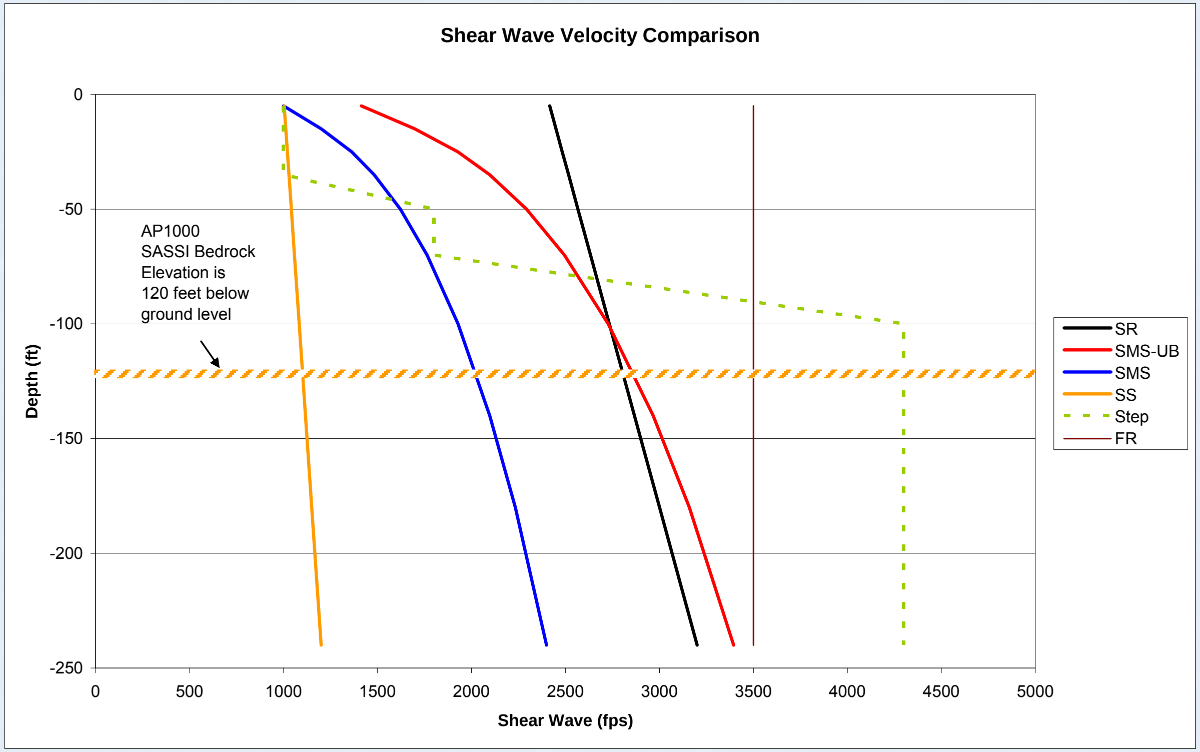
Source:
SSAR Appendix 2.5E, Table 5.1-1



Tank and Seismic Response Direction	Frequency Hertz
Fuel Area	
Fuel Pool, EW	0.39
Fuel Pool, NS	0.26
Fuel Transfer Canal, EW	0.68
Fuel Transfer Canal, NS	0.26
Cask Loading Pit, EW	0.39
Cask Loading Pit, NS	0.37
Cask Washdown Pit, EW	0.39
Cask Washdown Pit, NS	0.36
IRWST Tank	
Steel Wall, EW	0.41
Steel Wall, NS	0.25
NE Wall, EW	0.36
North Wall Pressurizer, NS	0.29
West Wall, EW	0.29
South Wall, NS	0.29
Shielding Building	
PCCS Tank	0.136

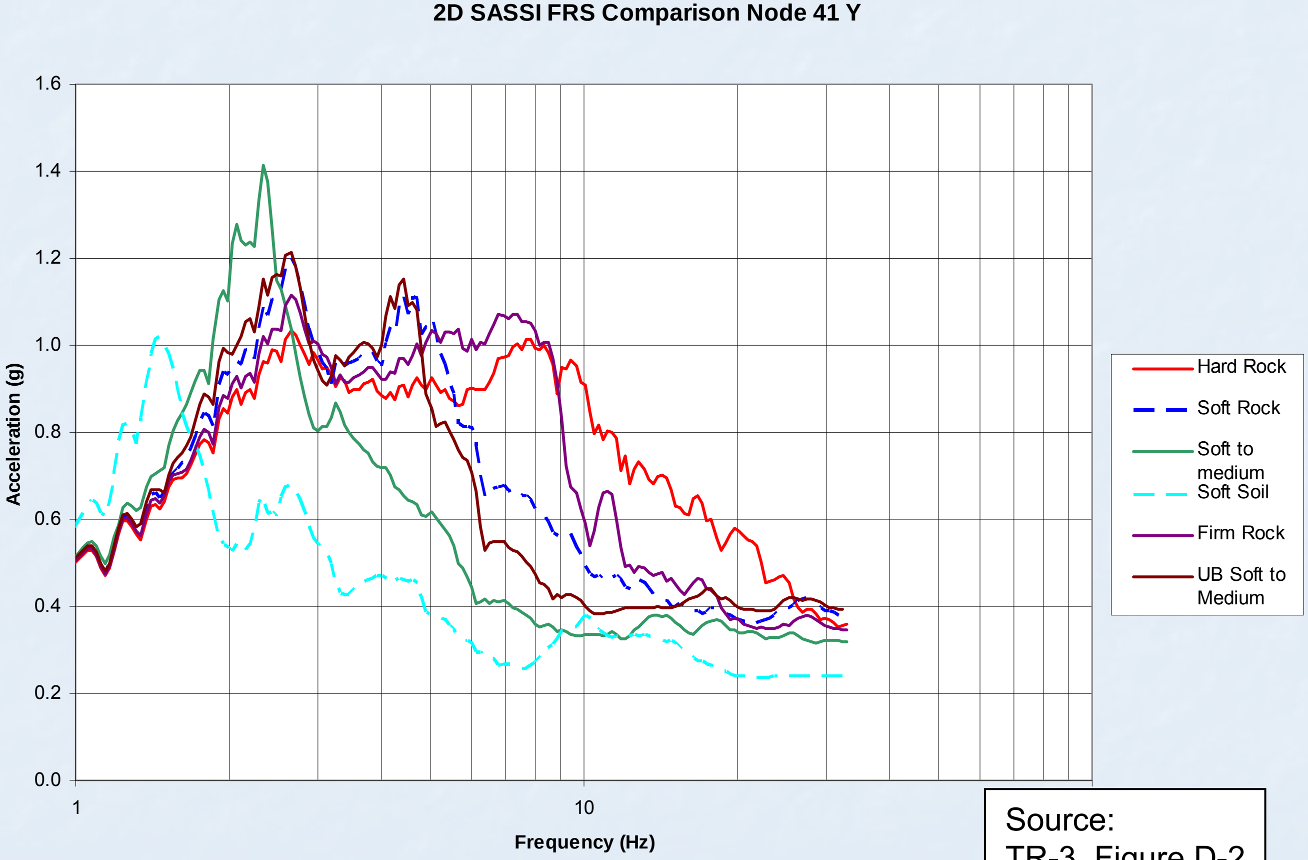
Source:
SSAR Appendix 2.5E, Table 5.1-2

AP1000 DCD



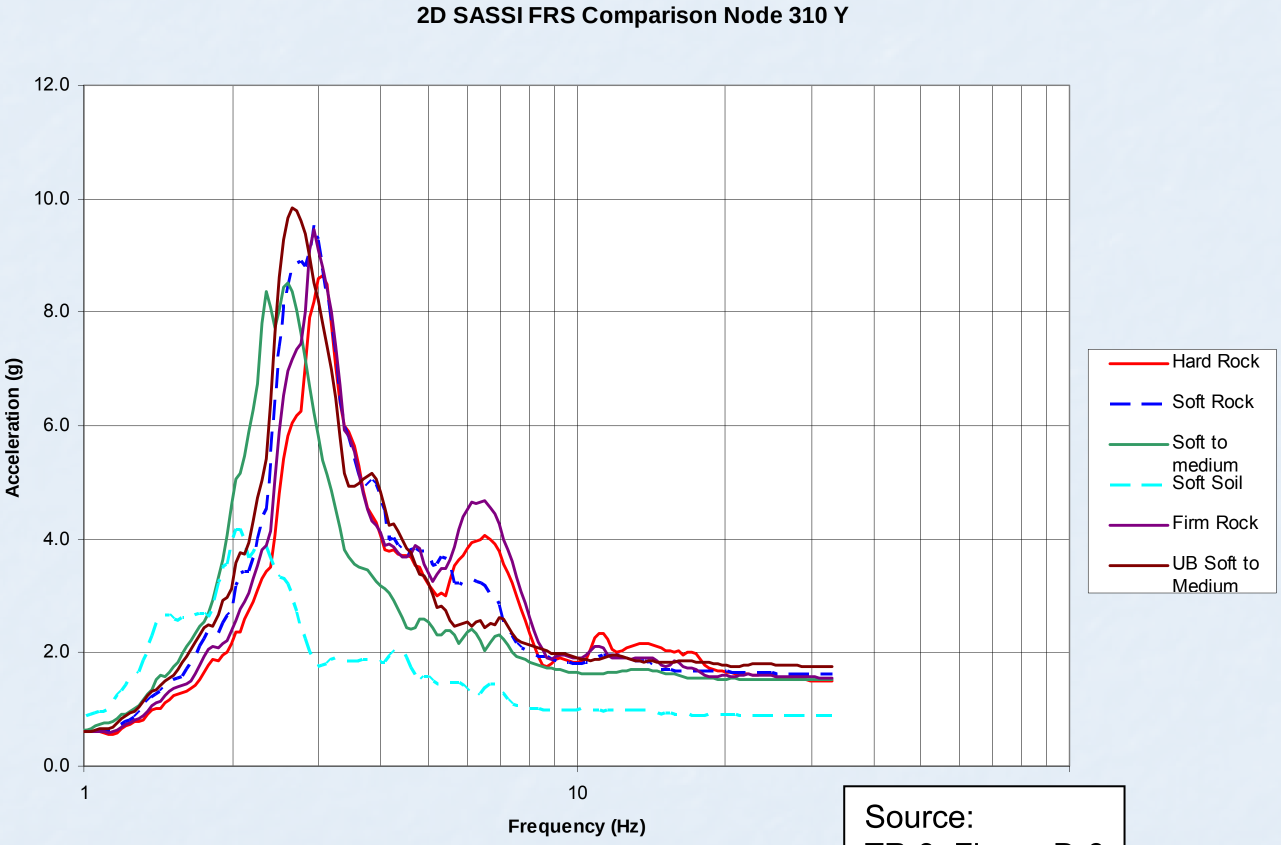
Source:
TR-3, Figure 4.4.1-1

AP1000 DCD



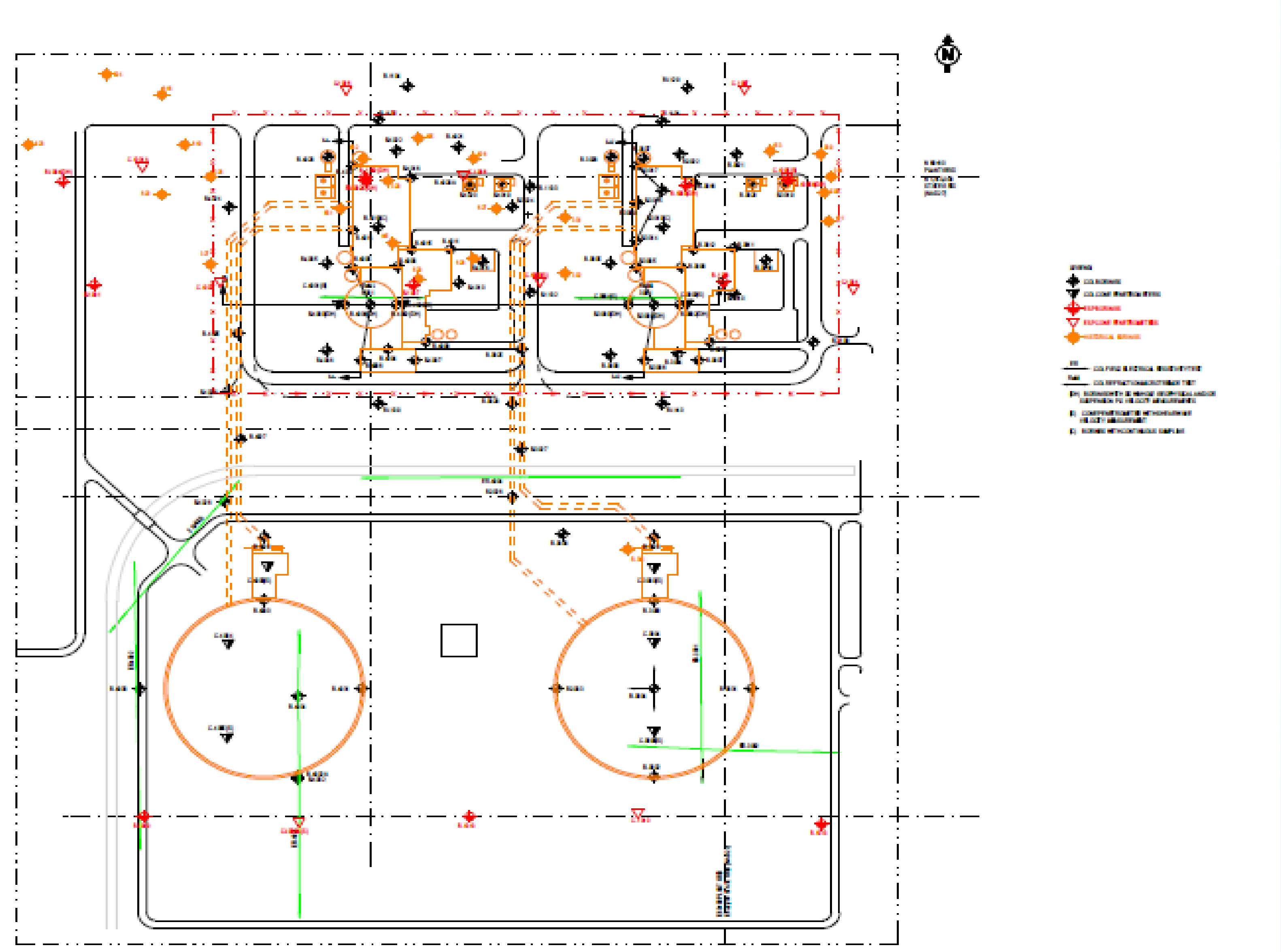
Source:
TR-3, Figure D-2

AP1000 DCD



Source:
TR-3, Figure D-6

COL Power Block – Cooling Tower Boring Locations



Southern Nuclear
Vogtle 3 & 4 ACRS Meeting
December 3-4, 2008

Early Site Permit

Jim Davis
ESP Project Engineer
Southern Nuclear

SOUTHERN COMPANY

Agenda

- Introduction
- Schedule
- Early Site Permit (ESP) Overview
- Limited Work Authorization (LWA) Overview

Introduction

- Southern Nuclear is pursuing an Early Site Permit (ESP) in accordance with 10 CFR 52 Subpart A-Early Site Permits
- In addition Southern Nuclear is seeking a Limited Work Authorization (LWA) in accordance with 10 CFR 50.10

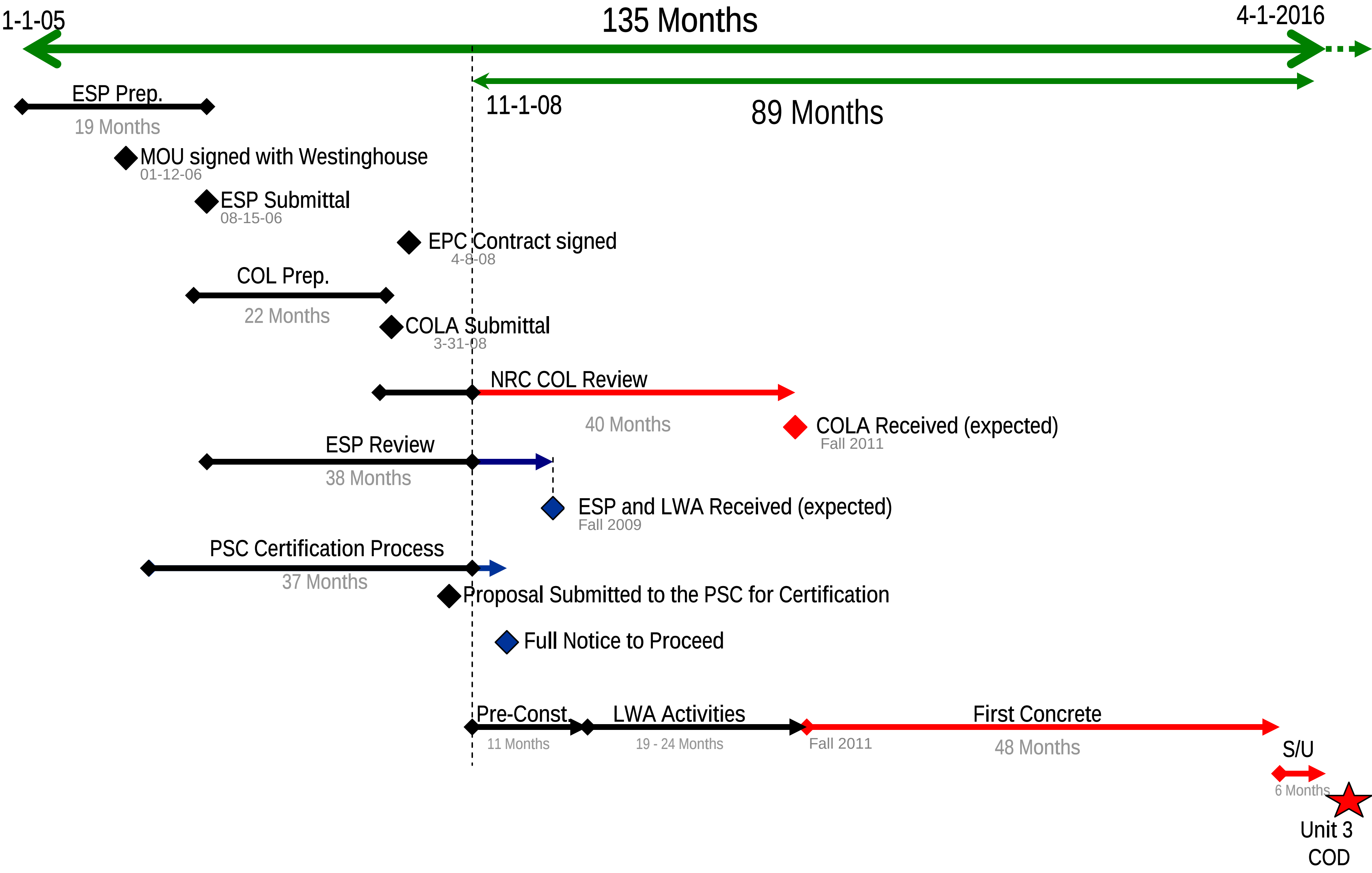
Introduction

- An ESP grants approval of a site for one or more nuclear power facilities separate from the filing of an application for a construction permit or combined license for the facility
- The requested LWA will allow a limited scope of safety-related construction activities to proceed at applicants risk as long as a site redress plan is included.

VEGP ESP Level of Detail

Example	Other ESPs	VEGP ESP
Reactor Type Power Output	Options Listed	Two Westinghouse AP1000’s at 1117 MWe Each
Plant Layout Cooling Water Design Intake Design	General Information Provided	Detailed Conceptual Design and Layouts Provided
Water Consumption And Discharge Flow	Envelope Approach	Plant-Specific Numbers Provided
Normal Effluents and Accident Doses	Envelope Approach	Plant-Specific Numbers Provided
Emergency Plan	Major Features	Complete & Integrated Plan
Limited Work Authorization	None	Requested for specific activities

Vogtle 3&4 Schedule



1-1-05

135 Months

4-1-2016

11-1-08

89 Months

ESP Prep.

19 Months

MOU signed with Westinghouse

01-12-06

ESP Submittal

08-15-06

EPC Contract signed

4-8-08

COL Prep.

22 Months

COLA Submittal

3-31-08

NRC COL Review

40 Months

COLA Received (expected)

Fall 2011

ESP Review

38 Months

ESP and LWA Received (expected)

Fall 2009

PSC Certification Process

37 Months

Proposal Submitted to the PSC for Certification

Full Notice to Proceed

Pre-Const.

11 Months

LWA Activities

19 - 24 Months

Fall 2011

First Concrete

48 Months

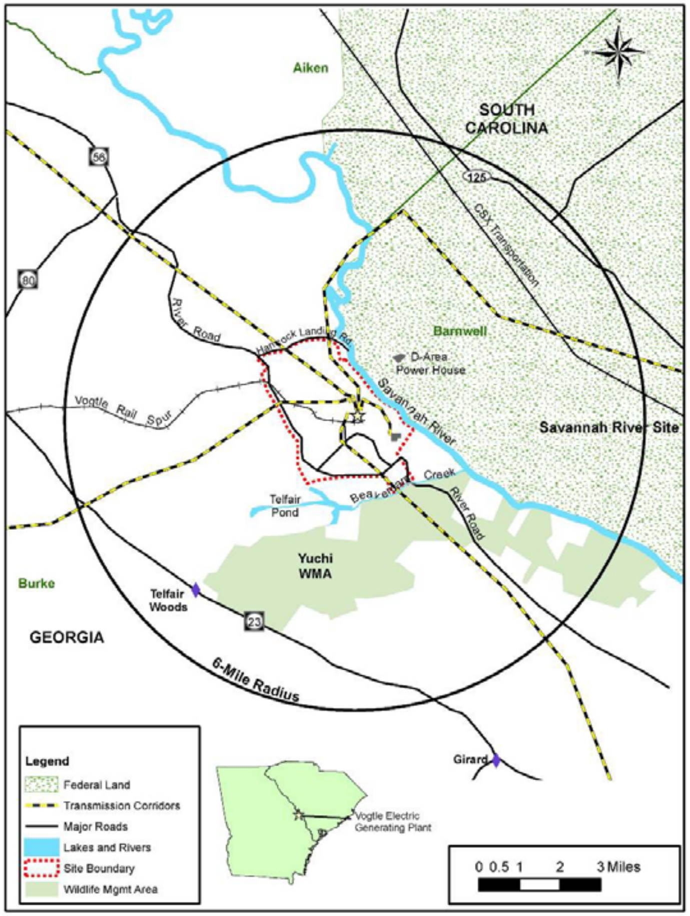
S/U

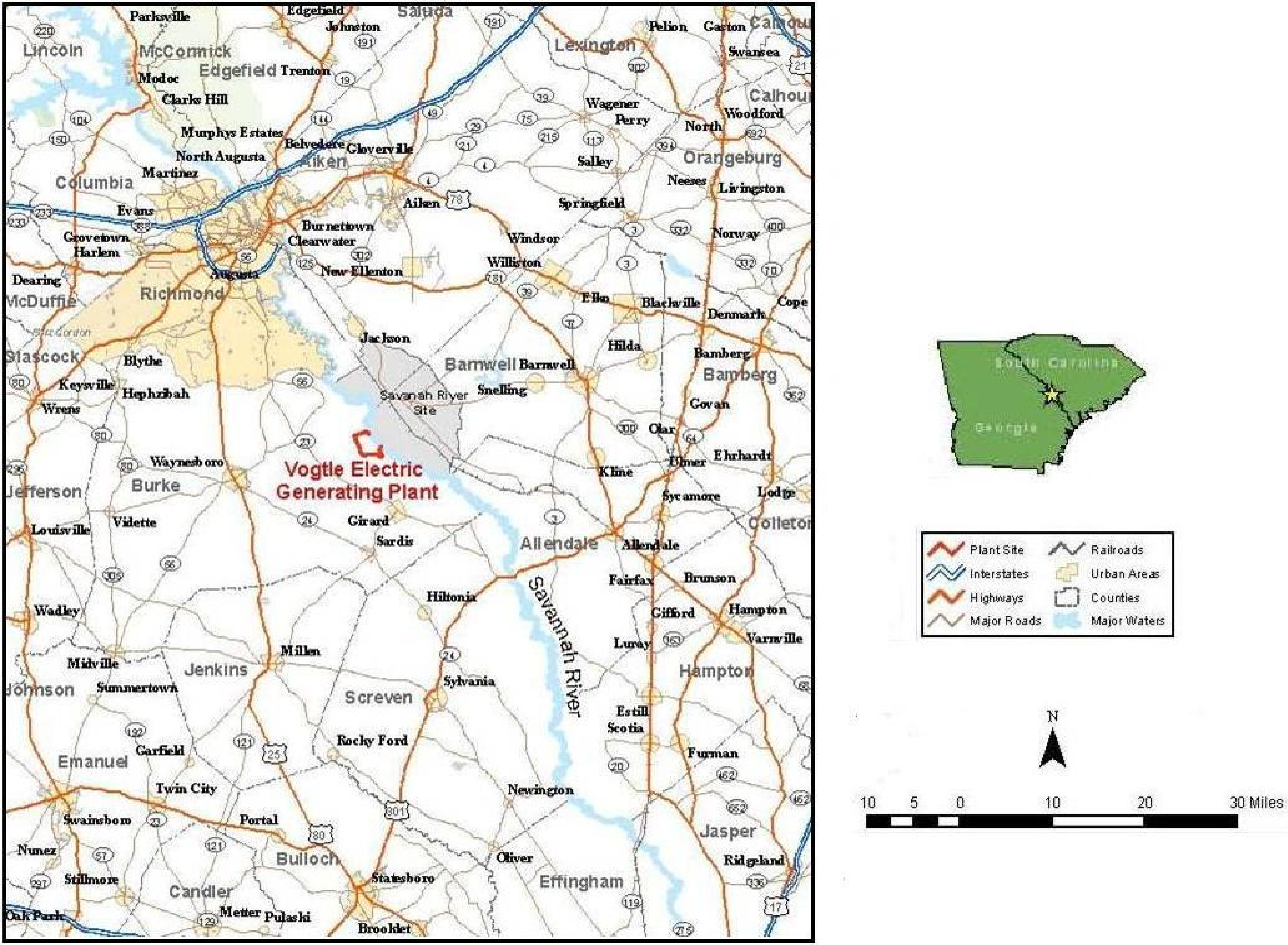
6 Months

Unit 3 COD

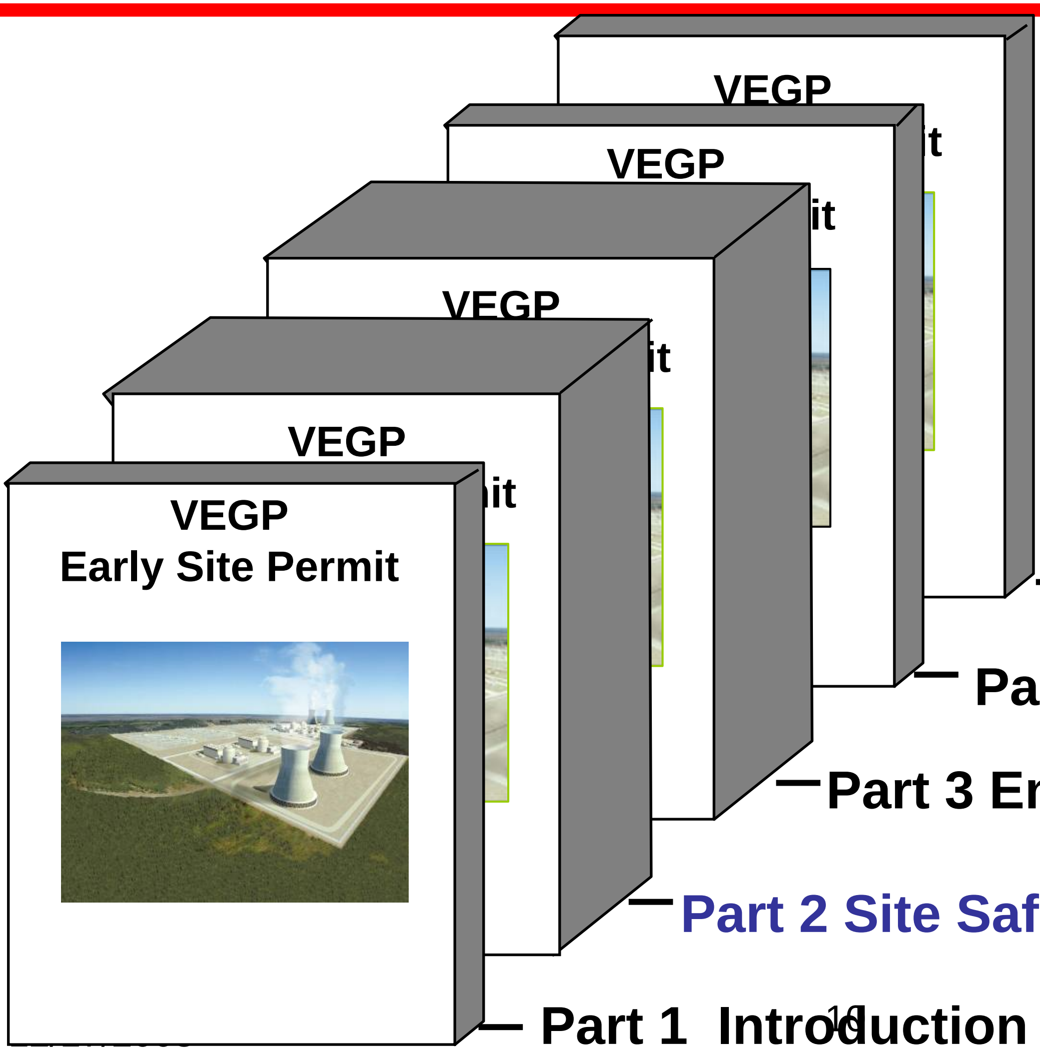
Vogtle Site Location

The, 3,169-acre existing 2 Unit site is located on a Coastal Plain bluff on the southwest side of the Savannah River in eastern Burke County Georgia. The site is directly across the river from the Department of Energy's Savannah River Site (Barnwell County, South Carolina). It is about 150 river miles from the mouth of the Savannah River and approximately 26 miles southeast of Augusta, Georgia.




Vogtle Electric Generating Plant
Savannah River Site
Savannah River
Parksville
Edgefield
Saluda
Johnston
Pelion
Gaston
Calhoun
Lincoln
McCormick
Edgefield
Trenton
Lexington
Swansea
Modoc
Clarks Hill
Wagener
Perry
Calhoun
Woodford
Murphys Estates
North
Belvedere
Gloverville
Salley
Orangeburg
North Augusta
Aiken
Martinez
Columbia
Neeses
Livingston
Evans
Aiken
Springfield
Burnettown
Clearwater
Windsor
Norway
Grovetown
Harlem
New Ellenton
Williston
Augusta
Dearing
McDuffie
Richmond
Elko
Blackville
Denmark
Cope
Jackson
Glascock
Blythe
Barnwell
Barnwell
Hilda
Bamberg
Bamberg
Keysville
Hephzibah
Snelling
Wrens
Govan
Olar
Waynesboro
Ehrhardt
Ulmer
Kline
Jefferson
Burke
Sycamore
Lodge
Colleton
Louisville
Vidette
Girard
Sardis
Allendale
Allendale
Fairfax
Brunson
Hiltonia
Wadley
Gifford
Hampton
Varnville
Luray
Midville
Millen
Jenkins
Hampton
Johnson
Summertown
Screven
Sylvania
Estill
Scotia
Garfield
Rocky Ford
Emanuel
Furman
Twin City
Newington
Swainsboro
Portal
Jasper
Nunez
Bulloch
Oliver
Ridgeland
Stillmore
Candler
Statesboro
Effingham
Metter
Pulaski
Oak Park
Brooklet
South Carolina
Georgia
Plant Site
Railroads
Interstates
Urban Areas
Highways
Counties
Major Roads
Major Waters
N
10
5
0
10
20
30 Miles

Early Site Permit (ESP) Contents



Part 5 Emergency Plan

Part 4 Redress Plan

Part 3 Environmental Report

Part 2 Site Safety Analysis Report (SSAR)

Part 1 Introduction

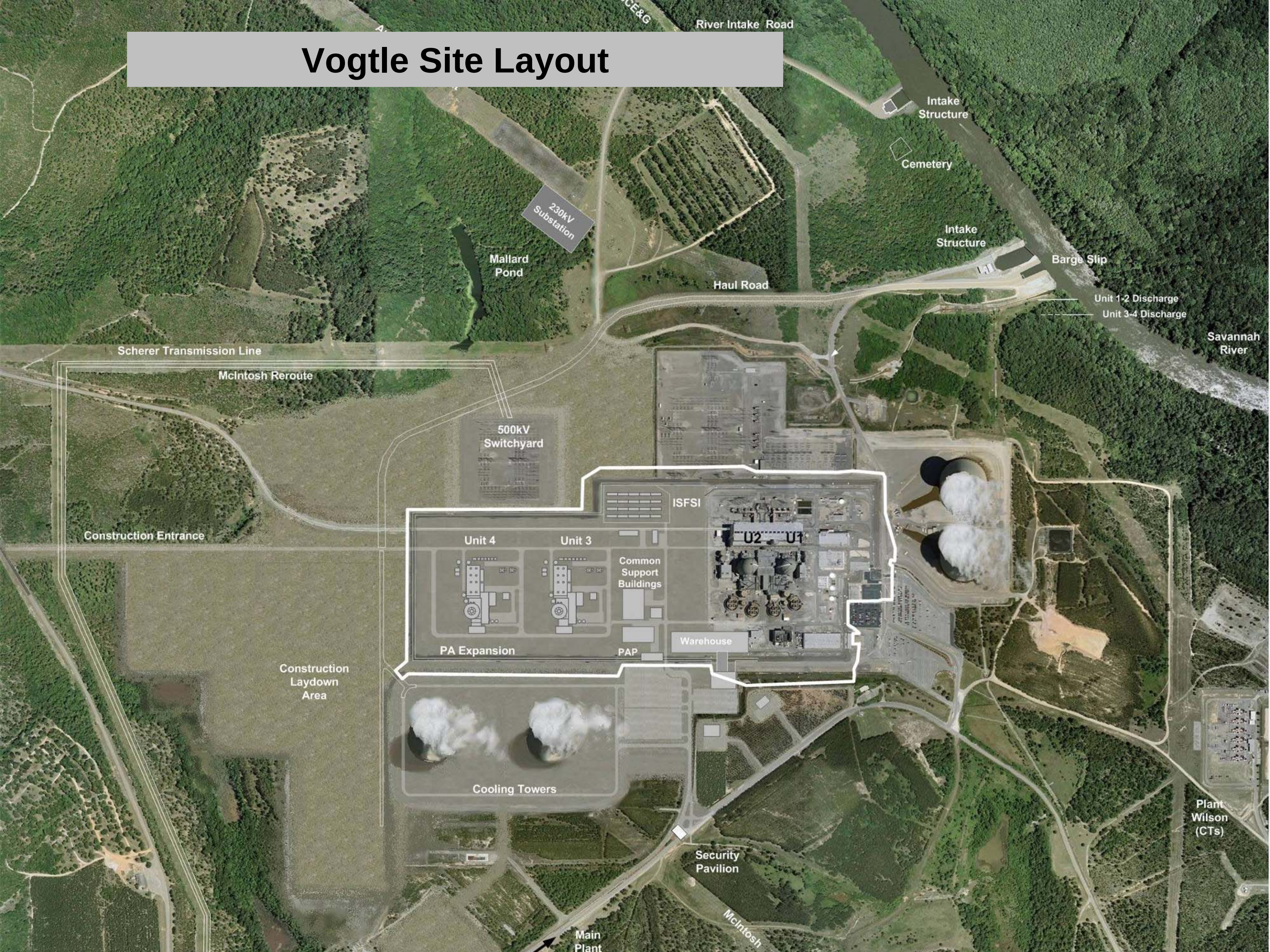
SOUTHERN COMPANY

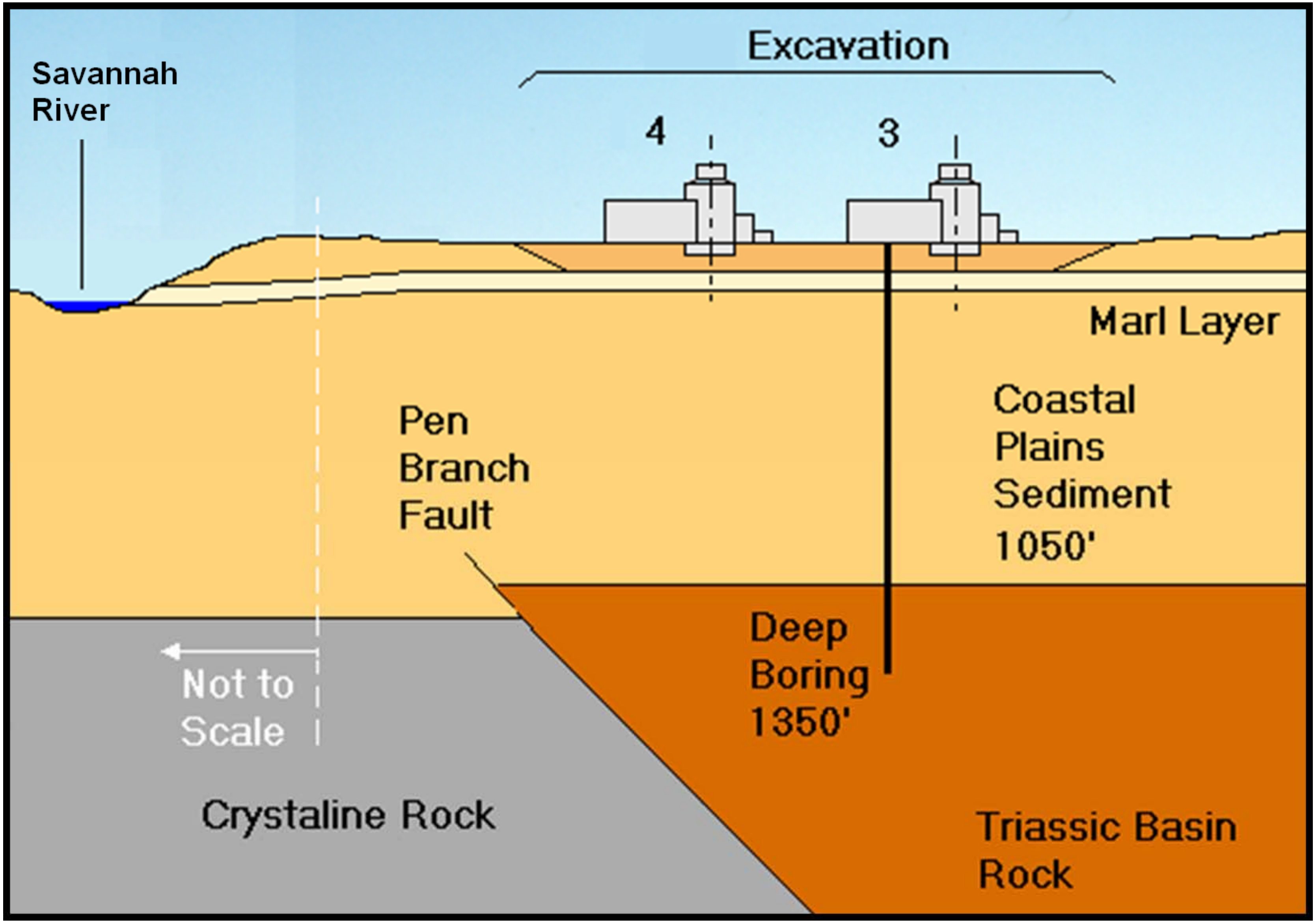
Part 2 Site Safety Analysis Report

Chapter numbering follows FSAR format and addressed selected chapters:

- 1 Introduction and General Description
- 2 Site Characteristics
  - 2.1 Geography and Demography
  - 2.2 Potential Hazards
  - 2.3 Meteorology
  - 2.4 Hydrology
  - 2.5 Geology and Seismic
- 3 Design of Structures, Components, Equipment, & Systems
  - 3.5.1.6 Aircraft Hazards
  - 3.8 Design of Category I Structures
- 11 Radioactive Waste Management
  - 11.2.3 Liquid Radioactive Releases
  - 11.3.3 Gaseous Radioactive Releases
- 13 Conduct of Operations
  - 13.3 Emergency Planning
  - 13.6 Industrial Security
  - 13.7 Fitness for Duty
- 15 Accident Analyses
- 17 Quality Assurance

Vogtle Site Layout





Site Soil/Rock Profile with Backfill

ESP Requests for Additional Information (RAIs)

Section	Subject	RAIs
2.1	Geography and Demography	12
2.2	Potential Hazards	18
2.3	Meteorology	16
2.4	Hydrology	10
2.5	Geology and Seismic	64
3.5.1.6	Aircraft Hazards	1
11	Liquid and Gaseous Releases	16
13	Emergency Planning	48
15	Accident Analysis	1
17	Quality Assurance	3
	RAIs	**189**

SER Open Items

Section	Subject	OIs
2.3	Meteorology	1
2.4	Hydrology	4
2.5	Geology and Seismic	22
13	Emergency Planning	13
	Total	**40**

LWA RAIs

The addition of the LWA request resulted in an additional 26 RAIs for the following subject areas:

- Site Investigation Information
- Engineering properties of subsurface materials
- Backfill requirements and engineering criteria

LWA and Preconstruction Overview

- Overview
- Pre-Construction Activities
- LWA Construction Activities
- LWA Schedule

Application Submittal - LWA

- Initial LWA-1 Request – ESP Revision 0, August 2006
- LWA-2 was included in ESP Revision 2, Supplement 1, August 2007
- Updated LWA Request to new rule 10 CFR 50.10 - ESP Revision 3, November 2007

Preconstruction Activities

Construction Does Not Include:

- Changes for temporary use of the land for public recreational purposes
- Site exploration
- Preparation of a site for construction of a facility
  - Clearing of the site
  - Grading
  - installation of drainage
  - Erosion and other environmental mitigation measures
  - Construction of temporary roads and borrow areas
- Erection of fences and other access control measures
- Excavation

Preconstruction Activities

Construction Does Not Include (Continued):

- Erection of support buildings for use in connection with the construction of the facility (Construction equipment storage sheds, Warehouse and shop facilities, Utilities, Concrete mixing plants, Docking and unloading facilities, Office buildings)
- Building of service facilities
- Paved roads
- Parking lots
- Railroad spurs
- Exterior utility and lighting systems
- Potable water systems
- Sanitary sewerage treatment facilities
- Transmission lines;
- Procurement or fabrication of components or portions of the proposed facility occurring at other than the final, in-place location at the facility

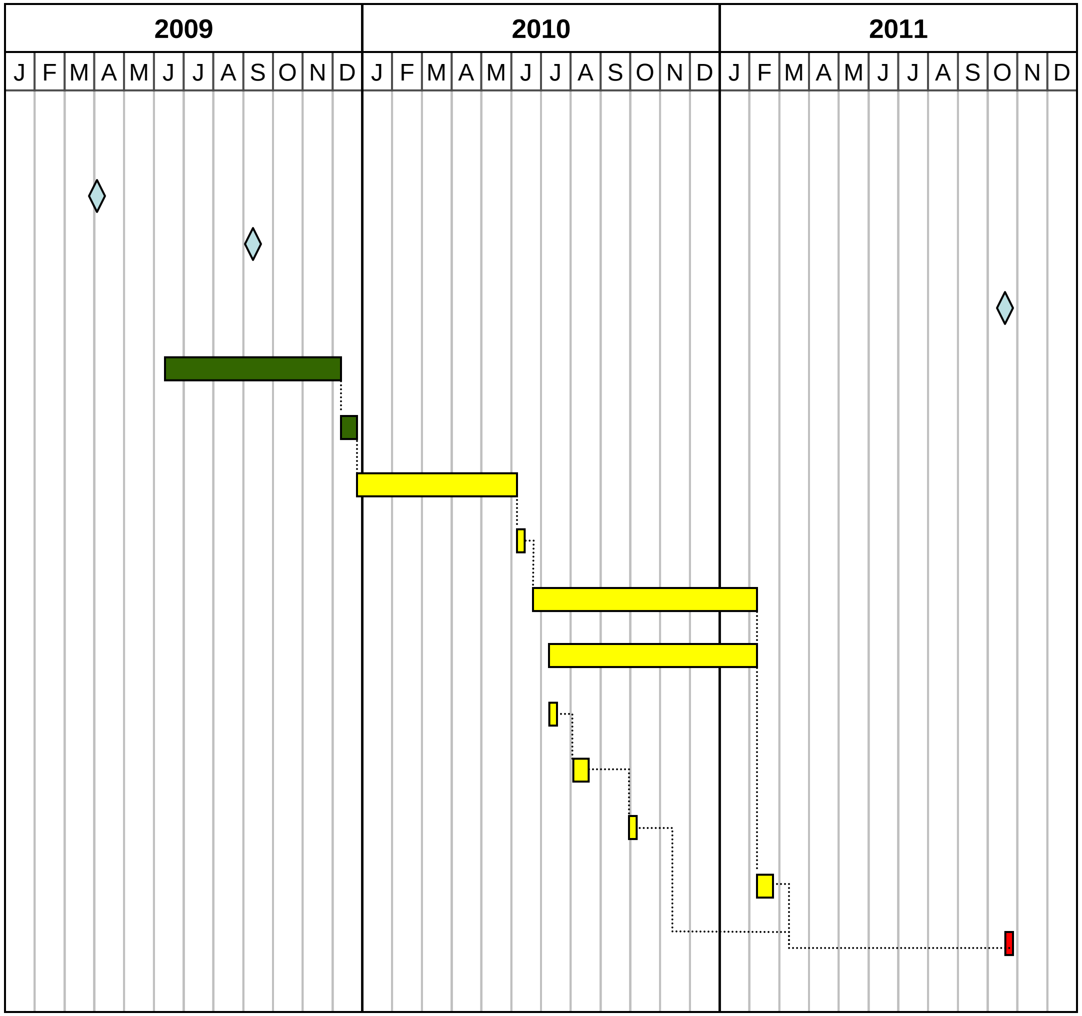
LWA Construction Activities

- The SNC LWA request is for the full extent of activities allowed by regulation and the site redress plan encompasses all such activities. Examples of VEGP LWA activities that SNC has identified include the following:
  - Engineered Backfill
  - Retaining Walls (mechanically stabilized earth walls)
  - Lean concrete backfill
  - Mud Mats
  - Waterproof membrane
  - FFD
  - QA
  - PI&R

SOUTHERN COMPANY

Vogtle Projected Construction Schedule
Activities Associated with LWA Request

Activity Description	Finish Date	Duration (weeks)
PSC Approval	4/2009	0
ESP Approval	9/2009	0
COL Approval	Fall 2011	0
Excavate power block	12/09	27
Perform geological mapping	12/09	3
Backfill to base of NI	6/10	24
Survey for MSE wall installation	6/09	1
Construct MSE wall to grade	2/11	30
Backfill to grade for Unit 3	2/11	30
Place first mudmat	7/10	1
Apply membrane to mudmat and wall	8/10	2
Place second mudmat	10/10	1
Apply waterproof membrane to wall	2/11	3
Place first concrete for Nuclear Island	Fall 2011	1



NOTES

All activities shown are for Unit 3. Unit 4 activities lag the Unit 3 activities and have a similar duration.

Schedule shown is based on LWA date of November 1, 2009 and COL date of June 30, 2011.

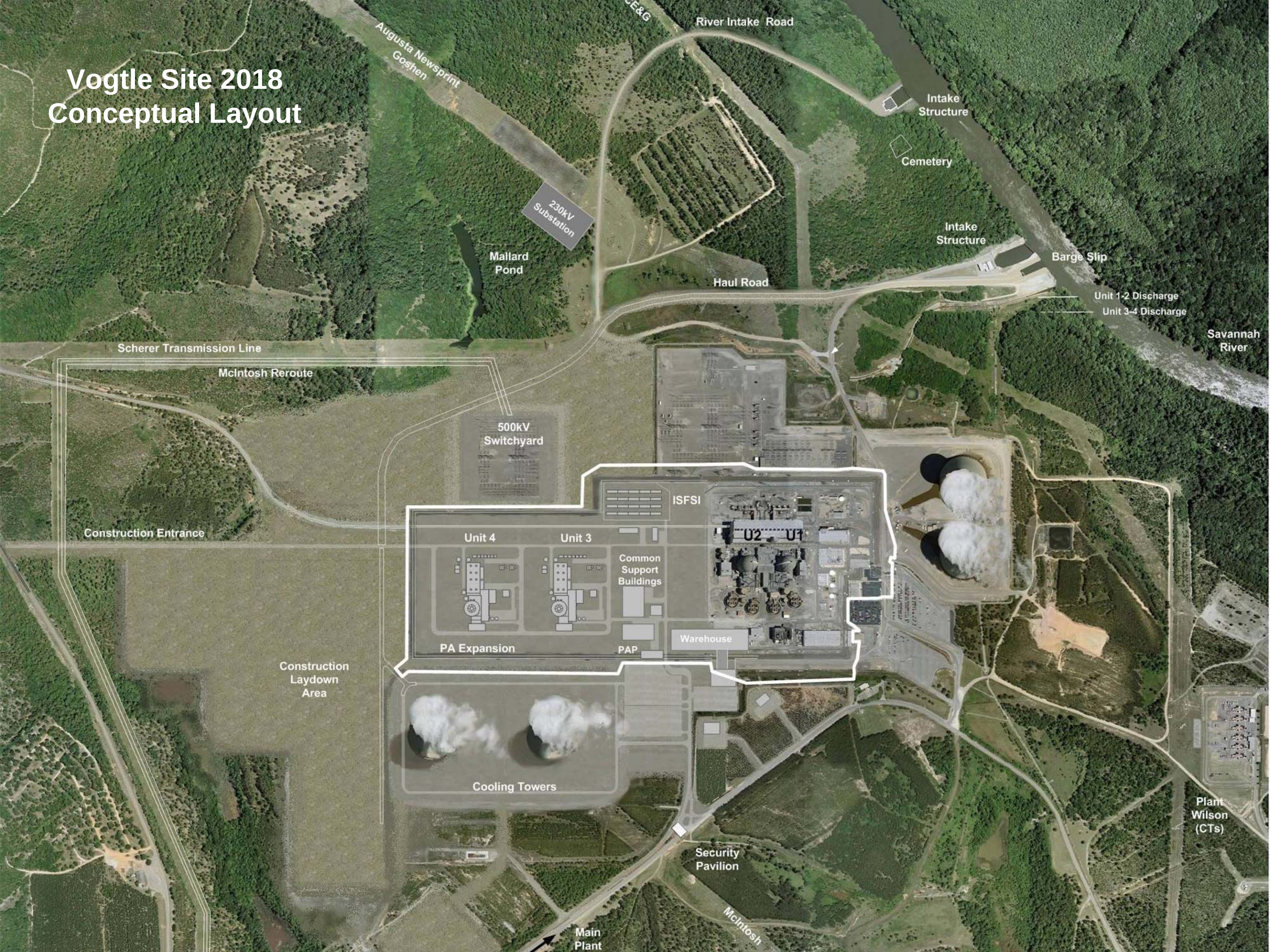
LEGEND

🟩 Pre-Construction Scope

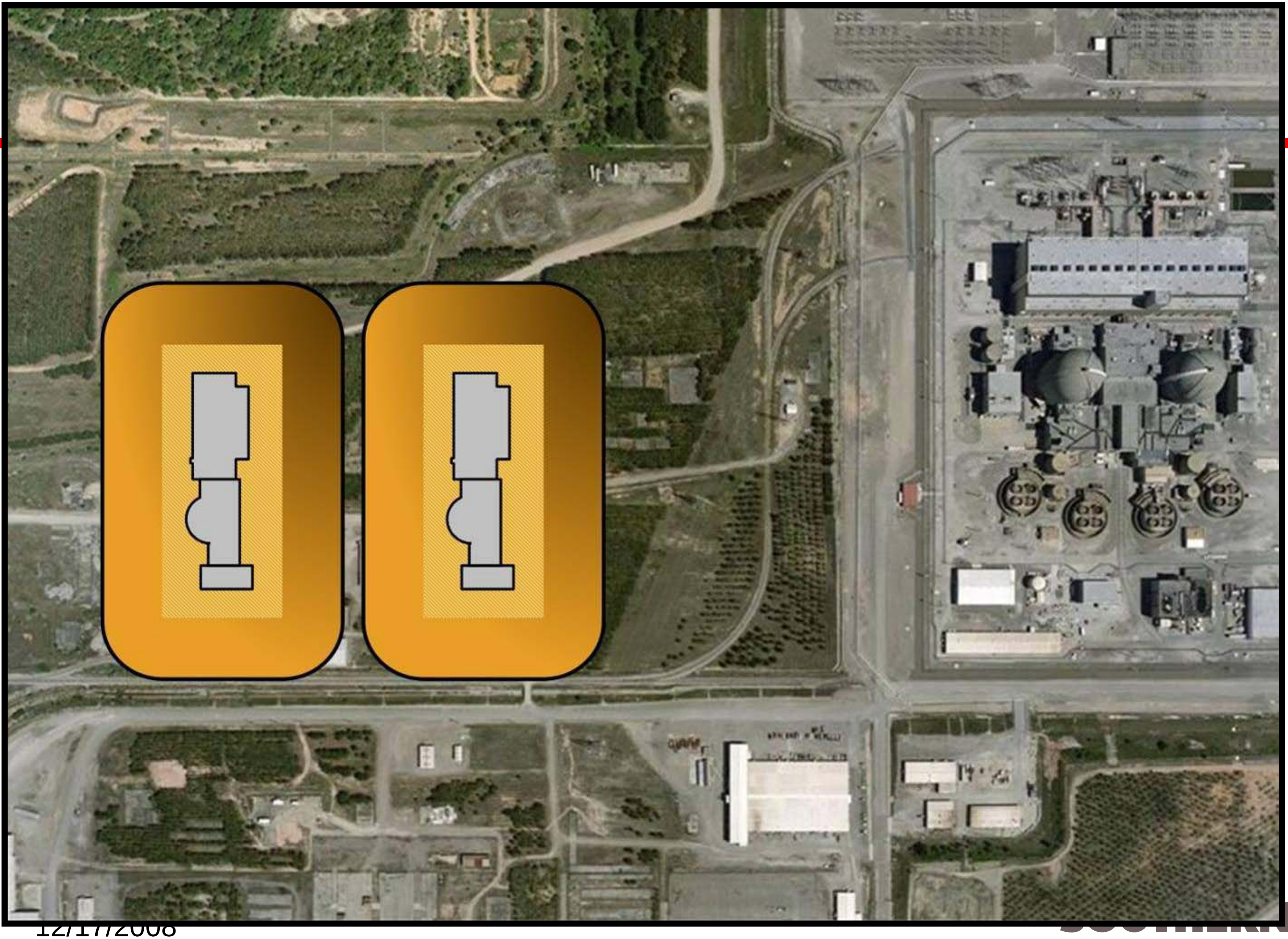
🟨 LWA Scope

🟥 COL Scope

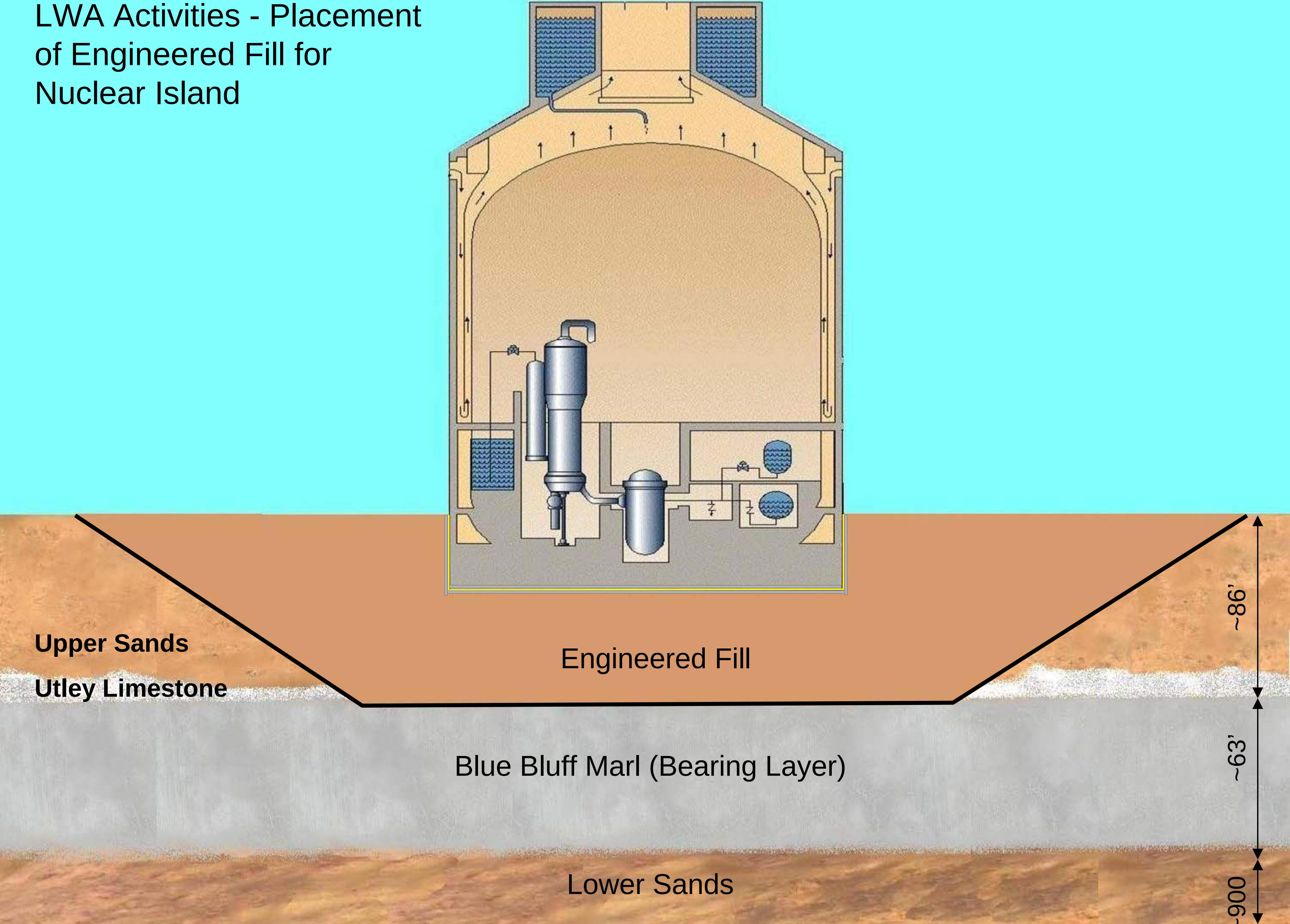
Vogtle Site 2018 Conceptual Layout

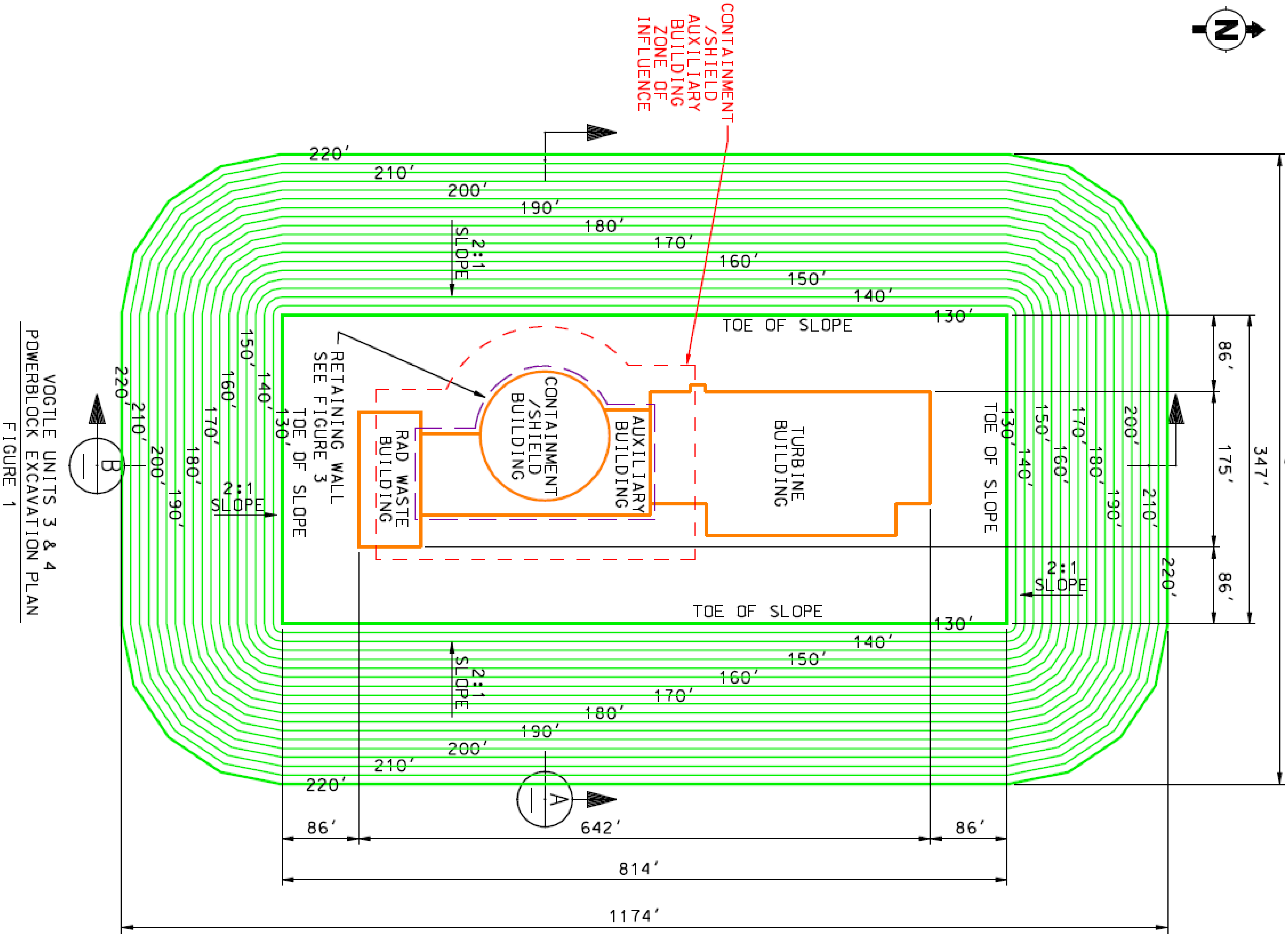


Preconstruction Activities - Dewatering and Excavation



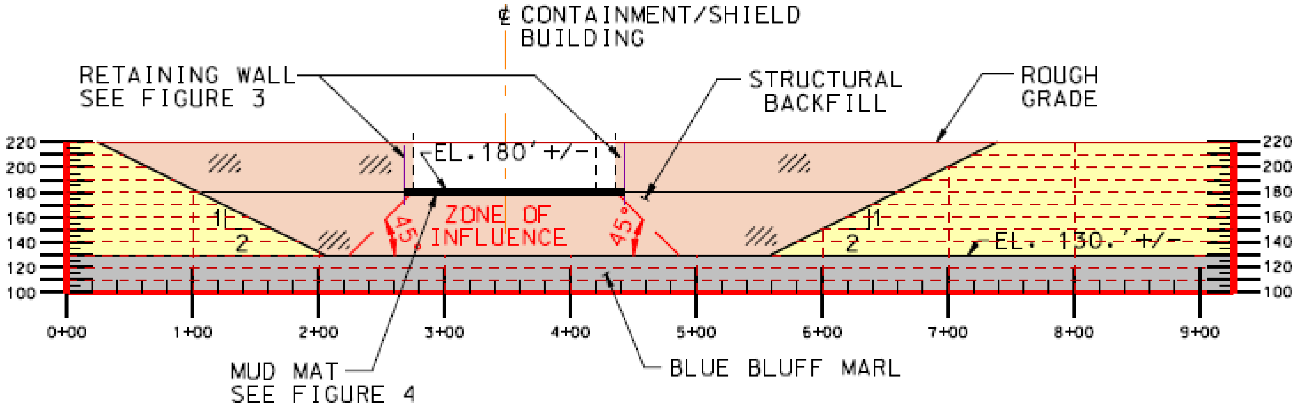
LWA Activities - Placement of Engineered Fill for Nuclear Island



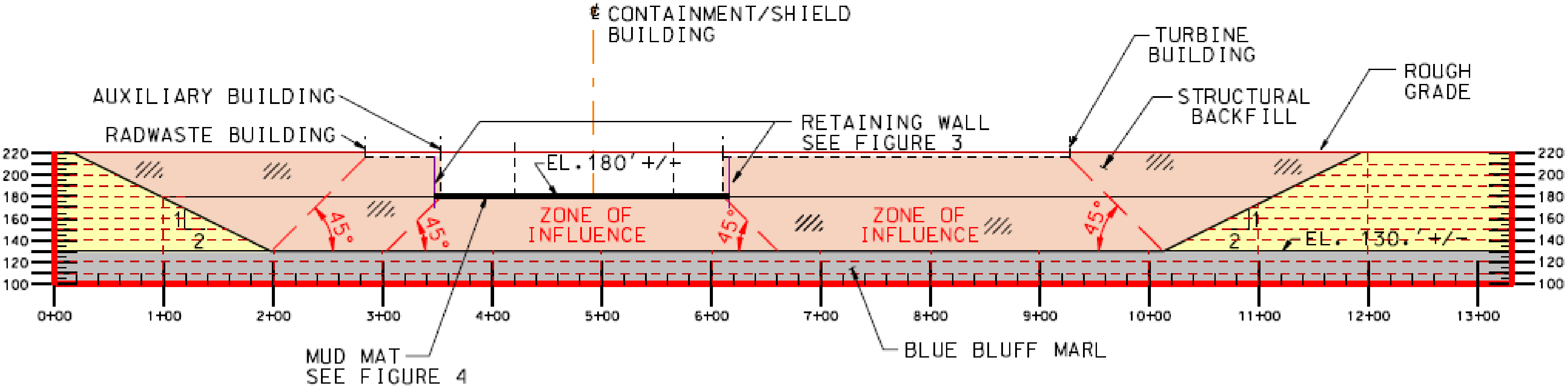


VOGTLE UNITS 3 & 4
POWERBLOCK EXCAVATION PLAN

FIGURE 1



SECTION A



SECTION B

VOGTLE UNITS 3 & 4
POWERBLOCK EXCAVATION SECTIONS
FIGURE 2

MSE Wall Test Section - July 2008

12/17/2008

Example MSE Wall near Atlanta Airport

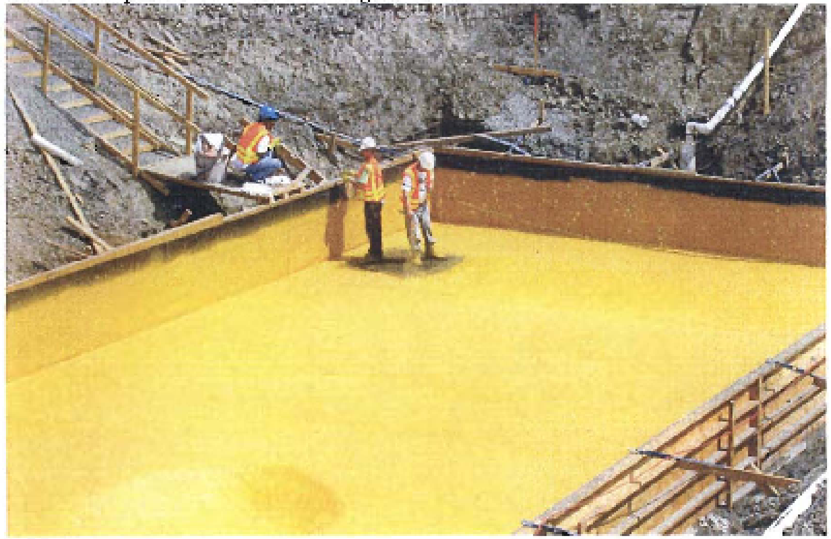
Waterproof Membrane

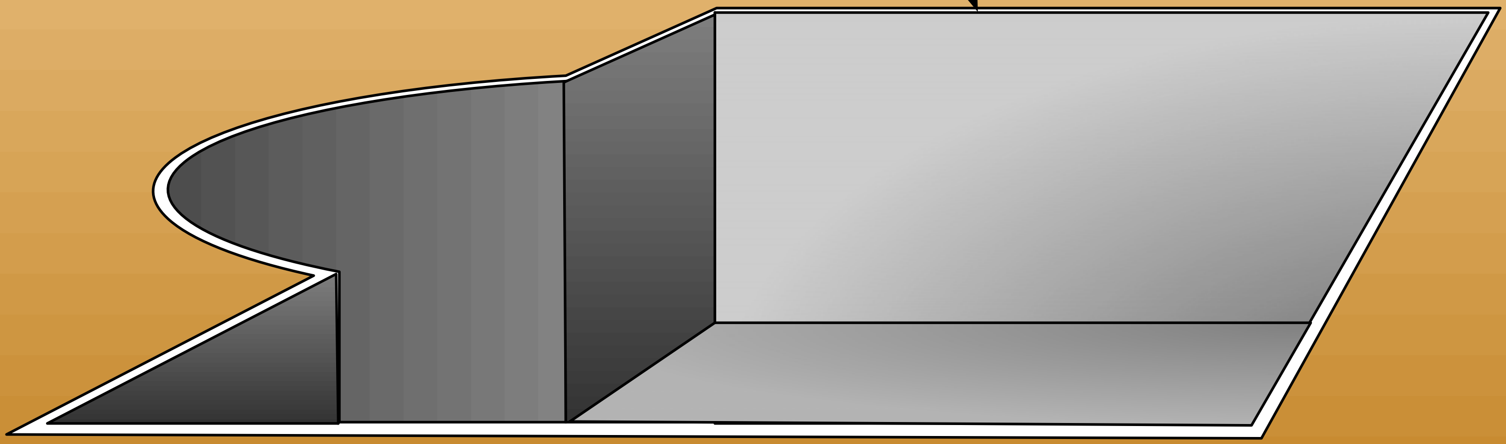
stirling lloyd
THE TECHNOLOGY OF PROTECTION

Installation

Above. Application of the 1st coat of Integritank (yellow) on to the geotextile. The walls are sprayed first, followed by the slab, particularly at smaller sites, to ensure the material is cured at the applicators' entrance and exit points.

Below. Completion of 1st coat of Integritank.

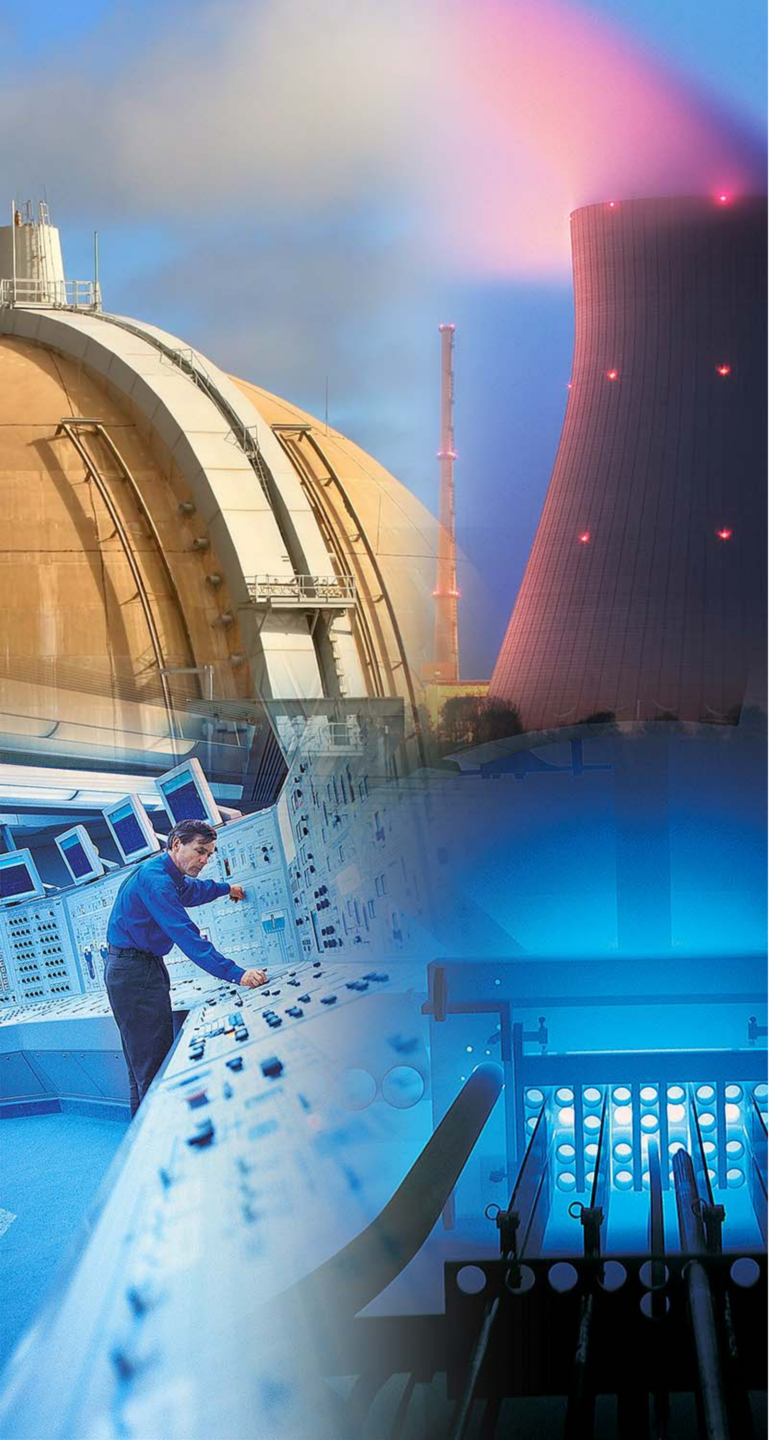




MSE Wall for Nuclear Island
In Place Prior to COL

Nuclear Island Foundation at Receipt of COL

Questions



ELECTRIC POWER
RESEARCH INSTITUTE

Industry Position Overview

On the Technical Basis for Revision of Embrittlement Criteria in 10CFR50.46

ACRS Committee Meeting
December 4, 2008
Rockville, MD

Industry Presentation Overview

Industry Collaboration with NRC

- The industry is supportive of NRC's overall objective with regards to revision of 10 CFR 50.46(b) to a performance based rule
- The Industry's Fuel Reliability Program (FRP) has been actively participating in the LOCA tests at ANL

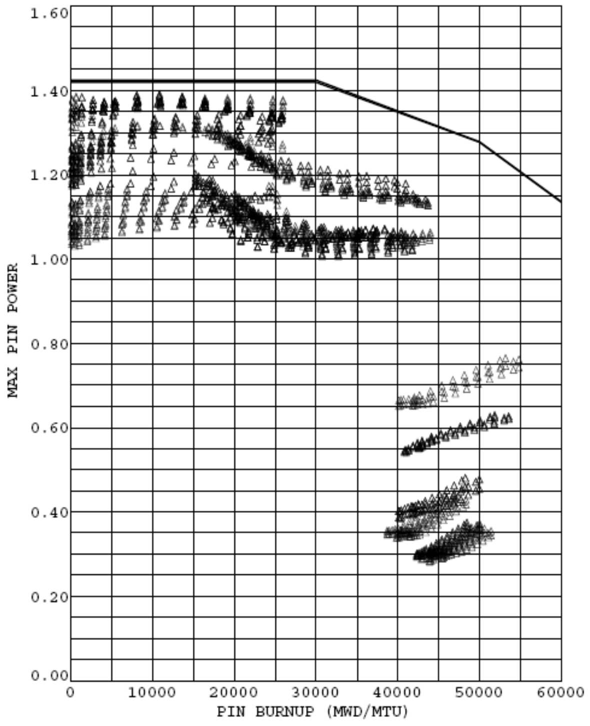
Industry Position Overview

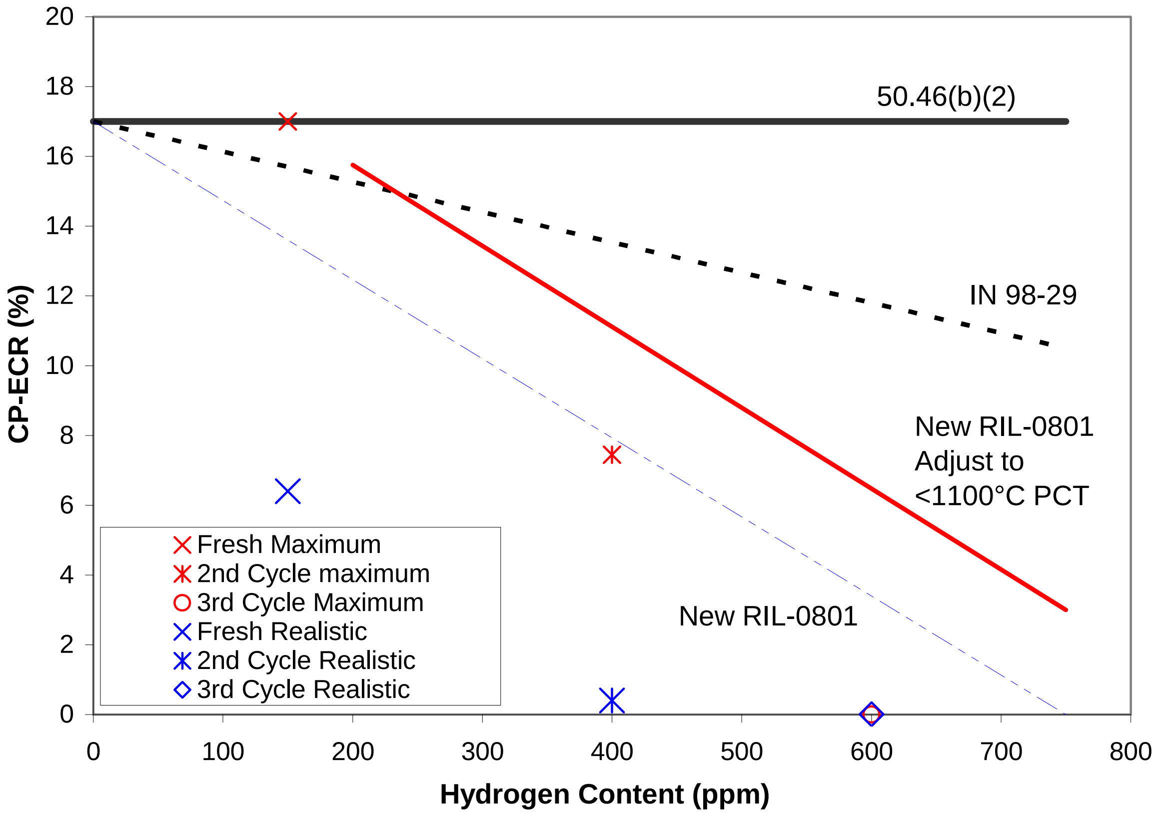
- Safety significance
- Industry position on hydrogen as a surrogate for irradiation
- Data gaps
- Estimate of implementation cost
- Summary

Safety Significance

- Evaluation indicates no significant safety concerns with respect to current design basis (based on typical Zircaloy-4 H pickup)

Typical 3-Loop PWR
Achievable Pin Power vs. Burnup

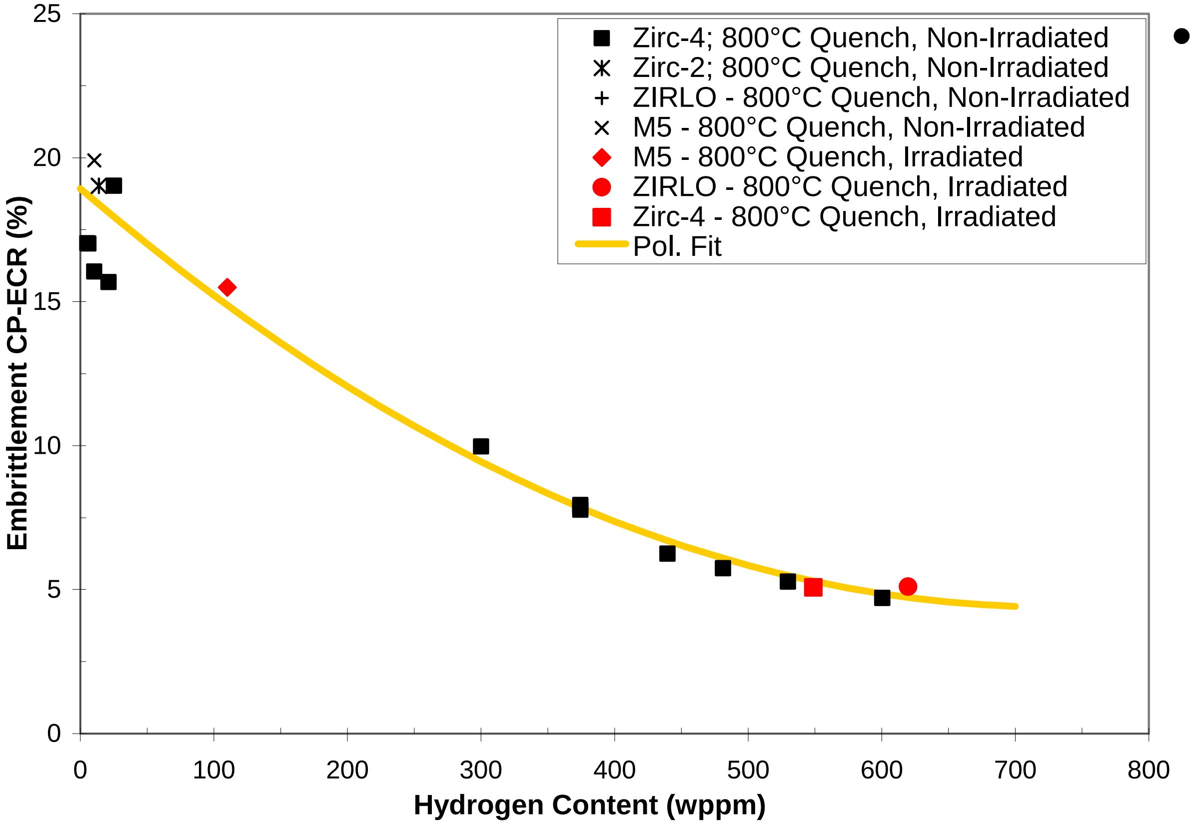




Industry Position on Hydrogen Pre-Charging

Pre-hydriding

- Zircaloy-4, ZIRLO and M5 PQD ductile-to-brittle transition ECR



- Sample from multiple vendors fit on the same trend line

Pre-hydriding appears to be a good surrogate for irradiation

Data Gaps

- More post-quench ductility data needed at lower temperatures with hydrogen effects
  - Industry is conducting complementary LOCA oxidation and PQD testing
- Requirement in RIL-0801 to use 2-sided oxidation away from the ballooned region is not supported by ANL data
  - Industry is planning to conducts tests to investigate potential influence of internal oxygen sources
- Periodic testing on breakaway oxidation is driven by observed short E110 breakaway time
  - Industry believes QA programs will be sufficient to keep process in control

Implications of Proposed Change - Cost Estimate

- Current LOCA evaluation models will likely require re-licensing
  - All operating reactors will need to demonstrate compliance
  - Expanded hot-cell campaigns to license corrosion-hydrogen correlations
- Costs to vendors and licensees to comply with anticipated new rule is estimated at several hundred million dollars
- Implementation will require multiple vendor/licensee/NRC interactions
  - Phased implementation a must if rulemaking proceeds

Summary

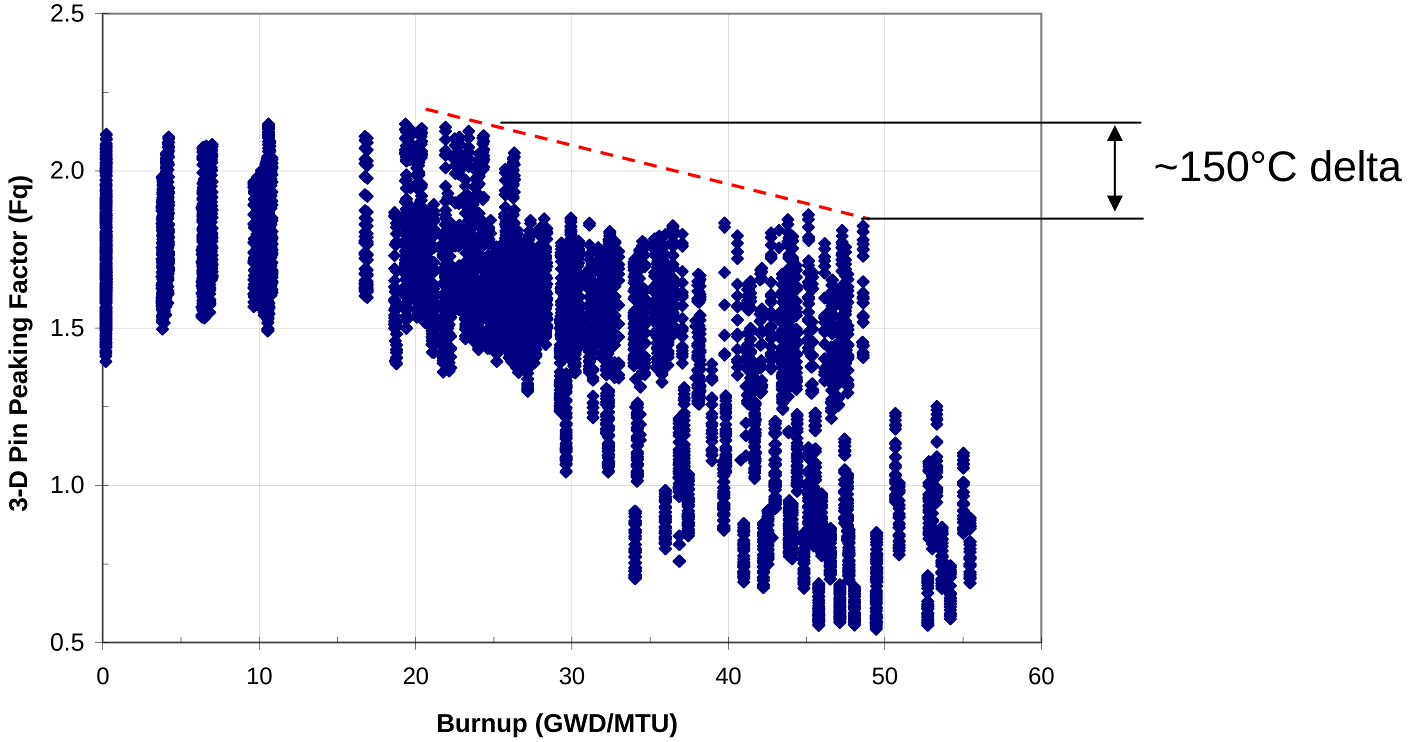
- Evaluation indicates no significant safety concerns with respect to current design basis, no need to rush through rule making
- Industry supports flexibility in rule
  - Use lower level documents for details
- Industry supports qualification testing, but not rule-mandated periodic testing
- Pre-hydriding appears to be a good surrogate for irradiation

Together…Shaping the Future of Electricity

Industry Position Overview

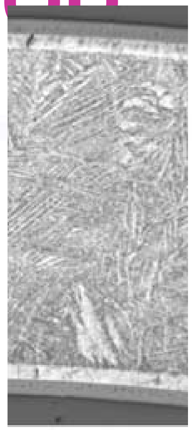
Incomplete database

– NRC-RES efforts focuses only on testing at 1200°C but high burnup fuel is not capable reaching this temperature



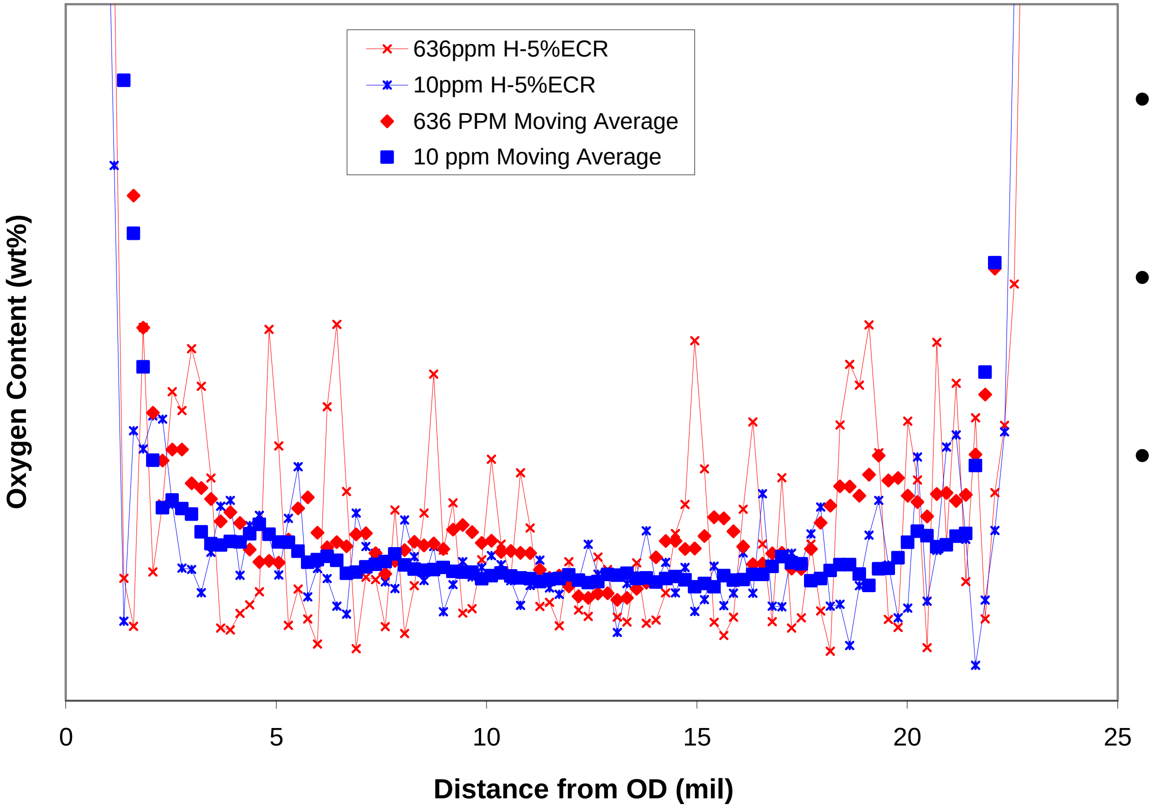
A bounding approach will have a significant negative impact on the industry with little or no safety benefit

Industry Test Plans – LOCA Oxidation & PQD



Preliminary evaluation example

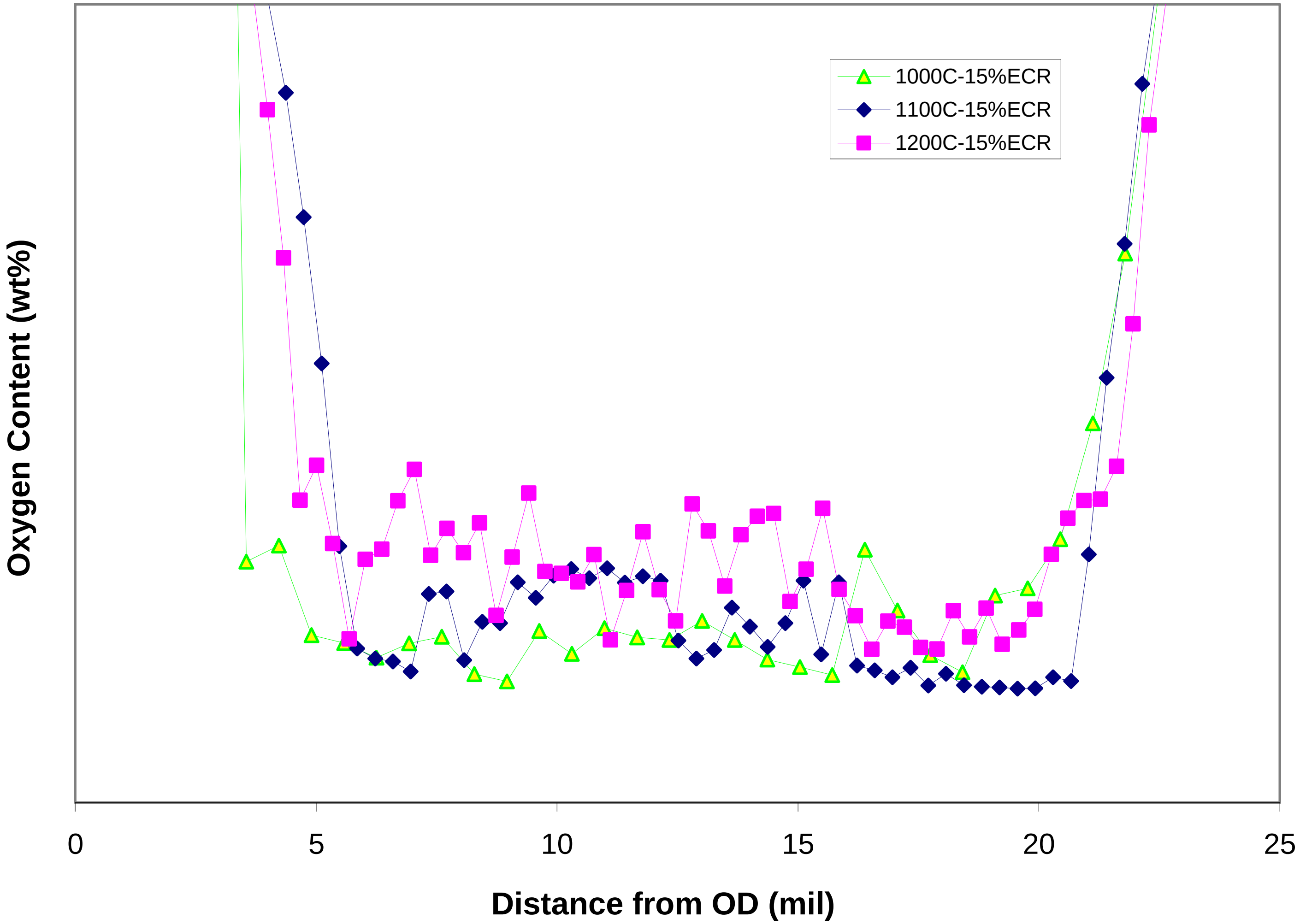
- Effect of hydrogen on oxygen content – actual ANL samples



- Hydrogen appears to enhance oxygen diffusion
- Hydrogen does not appear to increase oxygen solubility
- Other preliminary results indicate small increases in oxygen contents with increasing oxidation temperature

Industry Test Plans

- Oxygen content as a function of oxidation temperature

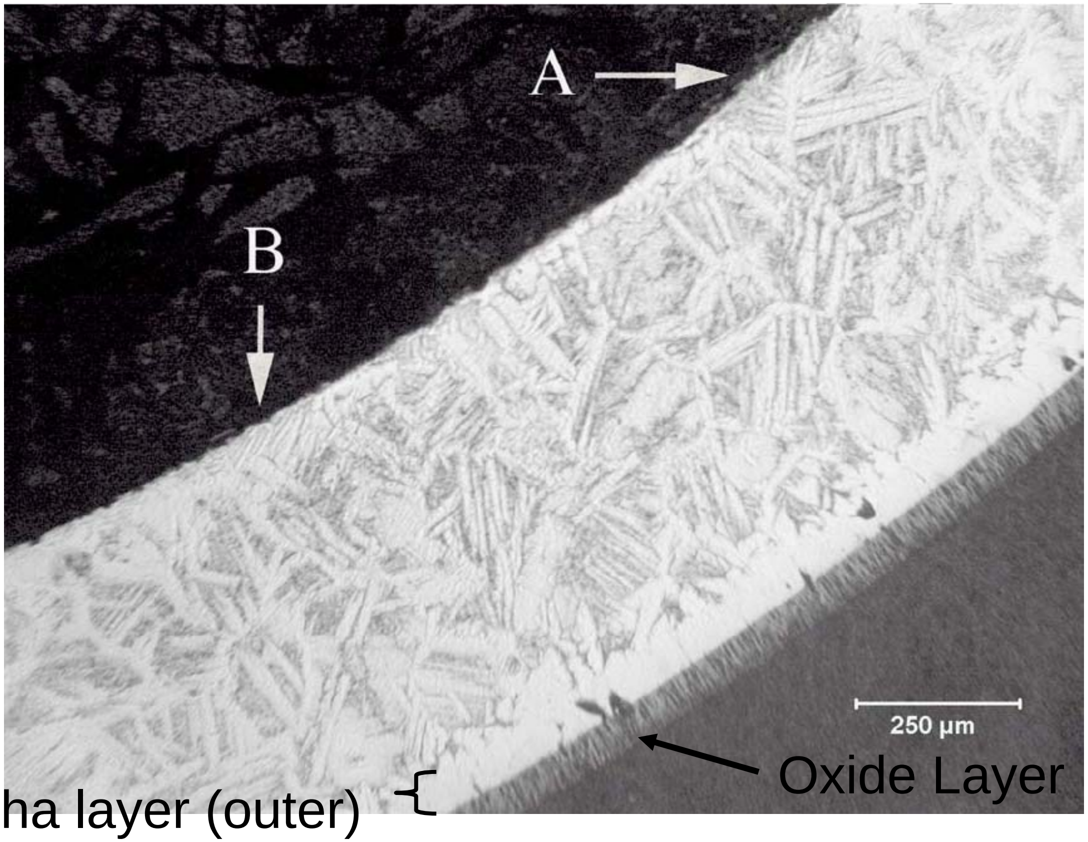


Integral LOCA Experiments on High-Burnup Limerick Fuel Do Not Show Significant ID Oxygen Pickup at 57 GWd/MTU Burnup

NUREG Interpretation

(A): This inner-surface region does not have an alpha layer even though in close proximity to fuel material. More typical of what is observed for most of the inner surface at this axial location.

(B): Evidence of some local inner surface oxygen-stabilized alpha



Oxide Layer

Alpha layer (outer)

Limerick fuel at 57-60 GWd/MTu

From Figure 175 : for ICL#2 sample at 50 mm above the burst midplane

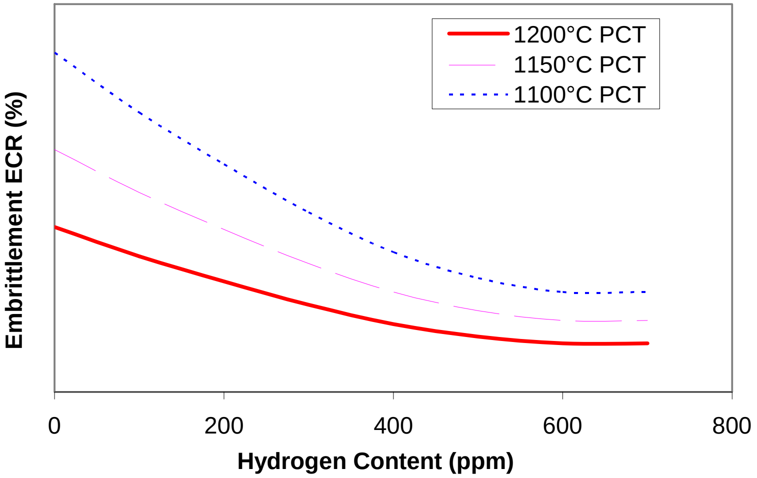
Industry Test Plans – LOCA Oxidation & PQD

Test Apparatus

- Electric resistance furnace
  - Center core quartz tube
  - Heat furnace up to set temperature and then insert samples
    - Initial inert gas atmosphere is possible
  - Excellent temperature repeatability
  - Cooling scenarios
    - Slow cool – turn off power and let furnace cool
    - Intermediate cool – remove quartz tube from furnace
    - Fast cool - remove quartz tube from furnace and forced cooling with fan
    - Quench – drop sample into a water tank
- Sample evaluation
  - RCT, Metallography, WDX and EDS with oxygen standards

Industry Test Plans – LOCA Oxidation & PQD

- Data indicates ECR accumulated at lower temperatures not as detrimental to ductility for Zircaloy-4
- Attempted to demonstrate ECR is not reduced to zero at elevated hydrogen contents
- Test details
  - Entire range of relevant oxidation temperatures and hydrogen content
  - Full characterization of samples in addition to RCT
- Generate sufficient test data to propose alternative PQD criteria not tied to 1200°C
- Determine feasibility of developing an embrittlement model



Industry Test Plans – ID Oxidation

ID Oxidation Test Matrix

- Sealed BWR cladding capsules with pellets
  - No or little contact pressure
    - Fresh/pre-oxidized cladding
    - Standard pellets, capsule pressure equal to atmosphere at temperature
  - Contact pressure
    - Fresh/pre-oxidized cladding
    - Oversized pellets, capsule evacuated
- Sealed BWR cladding capsule without pellet
  - Evaluate pre-existing ID oxide effect
    - Determine extent of ID oxygen stabilized alpha formation
  - Evaluate clad mechanical property at LOCA temperatures
    - Needed to evaluate contact pressure

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Strategy for Revising 50.46(b) Fuel Performance Criteria

ACRS Full Committee Meeting
December 4, 2008

Paul M. Clifford
Division of Safety Systems
Nuclear Reactor Regulation

Rulemaking Objectives

- Following Commission directive, develop a performance-based rule which enables licensees to use advanced cladding materials without needing an exemption.
  - Replace prescriptive criteria with performance-based regulatory requirements.
  - Expand applicability beyond “zircaloy or ZIRLO”.
- Capture results of High Burnup LOCA Research Program.
  - Research identified new embrittlement mechanisms which necessitate rule changes.

Applicability of Rule

Current Regulation:

- Paragraph (a)(1)(i) limits applicability to “zircaloy or ZIRLO”.

Research Finding:

- Empirical database includes wide range of zirconium alloys.

Plant Safety:

- No impact.

Strategy for Revising Regulation:

- Replace “zircaloy or ZIRLO” with less specific terminology (e.g., approved zirconium-alloy).
- Applicability to new alloys will need to be demonstrated by testing.

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Peak Cladding Temperature

Current Regulation:

- Paragraph (b)(1) limits PCT to 2200☜F.

Research Finding:

- Post quench ductility (PQD) decreases dramatically in samples oxidized beyond 2200☜F.
- Confirms current regulatory criterion.

Plant Safety:

- No impact.

Strategy for Revising Regulation:

- No change.

Local Oxidation

Current Regulation:

- Paragraph (b)(2) limits local oxidation to 17% ECR.

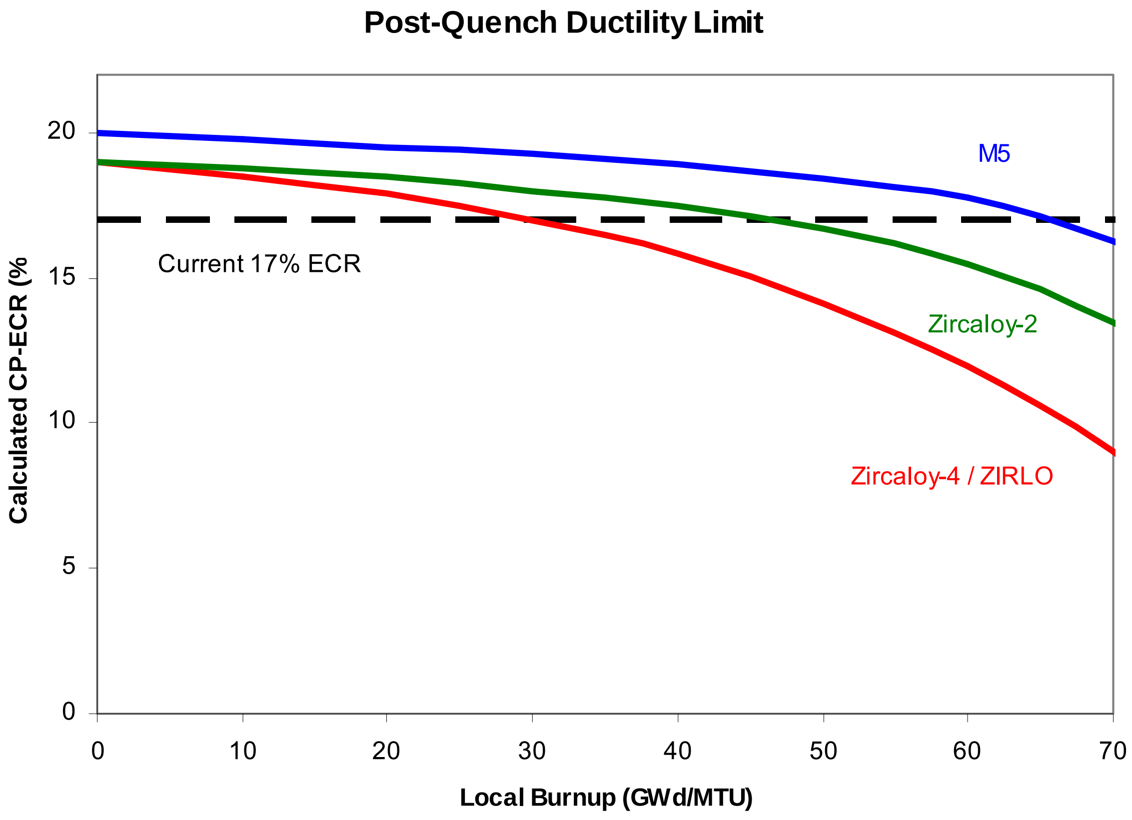
Research Finding:

- New cladding embrittlement mechanism identified.
    - PQD sensitive to pre-transient cladding hydrogen concentration.
- A constant 17% ECR limit does not always ensure PQD.
- Information Notice 98-29 adjustment (subtract initial oxide layer from 17% ECR limit) may not always ensure PQD.

Local Oxidation (cont.)

Plant Safety:

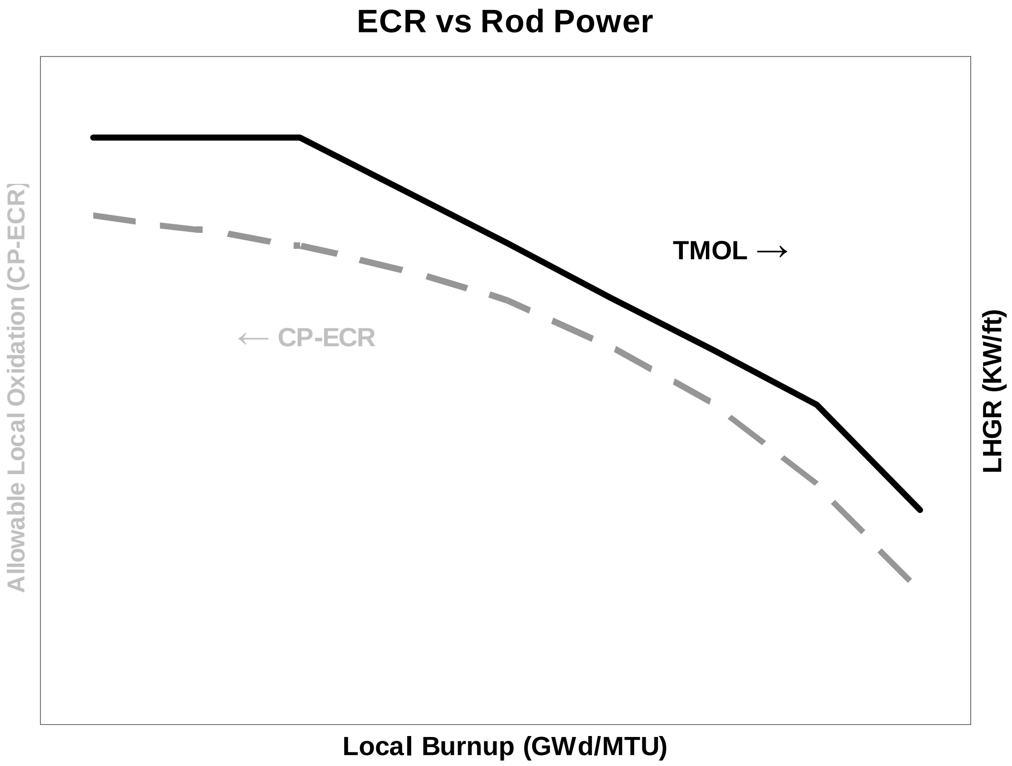
- Modern alloys exhibit unirradiated brittle transition at or above 17% ECR.



Local Oxidation (cont.)

Plant Safety:

- Highest power fuel rods challenge 2200☜F and 17% ECR limits.
- Corrosion build-up coincident with U^{235} depletion (diminishing rod power).
- Lower power fuel rods experience more benign transient.

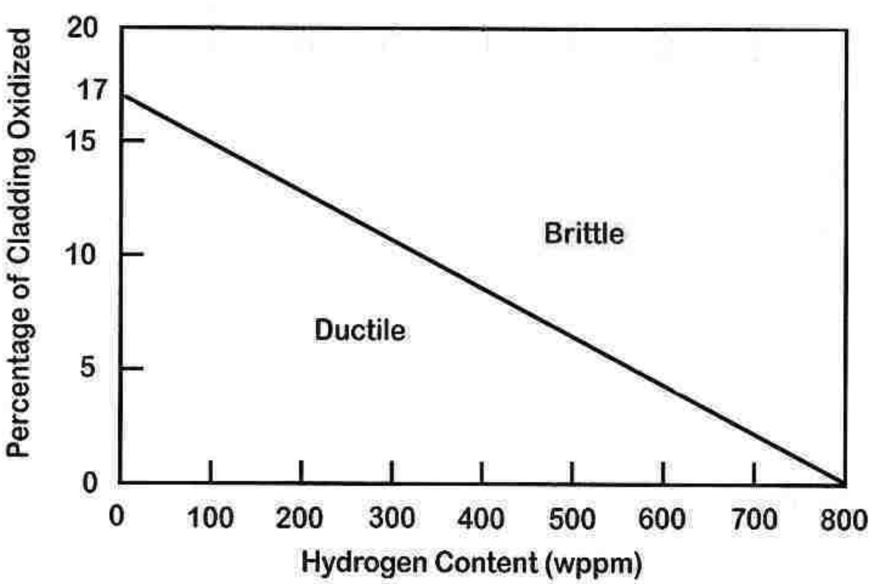


Local Oxidation (cont.)

Strategy for Revising Regulation:

Alternative Regulations:

1. Generic PQD criteria specified within rule.



2. Optional test program for defining alloy-specific or temperature-specific PQD criteria.

ID Oxygen Diffusion

Current Regulation:

- None.

Research Finding:

- Oxygen from fuel bonding layer (on cladding ID) diffuses into the base metal and exacerbates cladding embrittlement.

Plant Safety:

- Current methods require double sided oxidation within the balloon region.
- Higher burnup fuel rods operating at lower power will experience more benign transient.

Strategy for Revising Regulation:

- New requirement within rule.

Breakaway Oxidation

Current Regulation:

- None.

Research Finding:

- New cladding embrittlement mechanism identified.
  - Protective tetragonal oxide transforms to monoclinic structure.
  - Hydrogen uptake promotes cladding embrittlement.
- Timing of transformation sensitive to manufacturing process.

Plant Safety:

- Measured breakaway time for domestic alloys exceed 3000 seconds.
- SBLOCA analysis coupled with reasonable operator actions show that the duration at elevated temperatures remains below breakaway time.

Breakaway Oxidation (cont)

Strategy for Revising Regulation:

- New performance requirement within rule.
  - Required testing to establish measured break-away time.
- Required periodic testing.

Regulatory Challenge

- Developing a performance-based rule which meets the objectives of the rulemaking plan (e.g., optional testing program) while satisfying legal requirements (e.g., specific enforceable requirements).
  - Performance-based rule more difficult to script.
  - Specifying optional test protocols within rule versus regulatory guidance document.

Optional Test Program

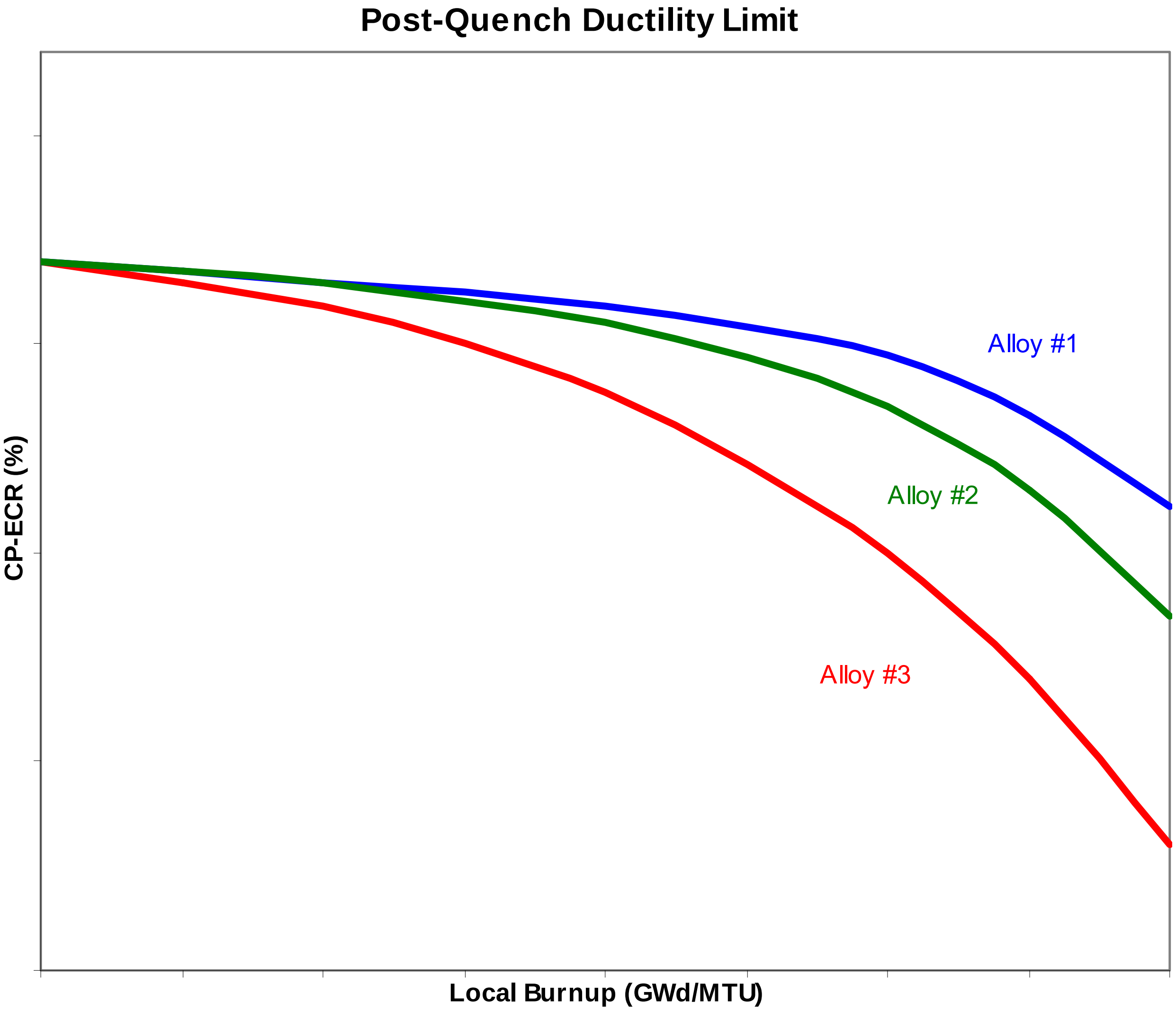
- Regulations within 50.46(b)(2) specify general requirements for optional testing:
  - Criterion for the ductility test would be 1% plastic strain using ring-compression tests.
  - Criterion for the breakaway oxidation test would be 200 wppm hydrogen uptake.

- Acceptable experimental protocols for establishing cladding ductility criteria and breakaway oxidation limits would be provided within a comprehensive test procedure.

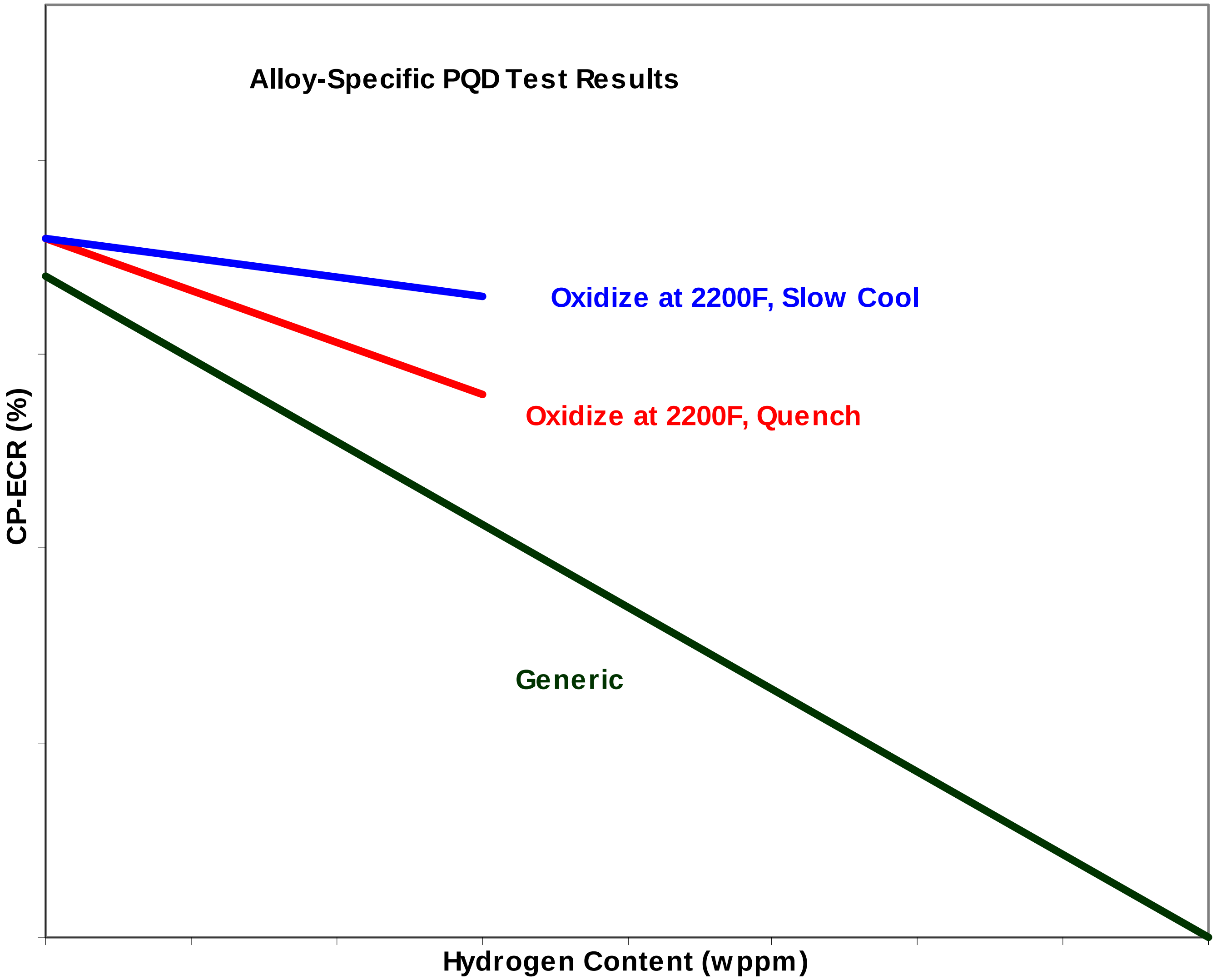
Implementing Alternative PQD Criteria

Implementing PQD Curve

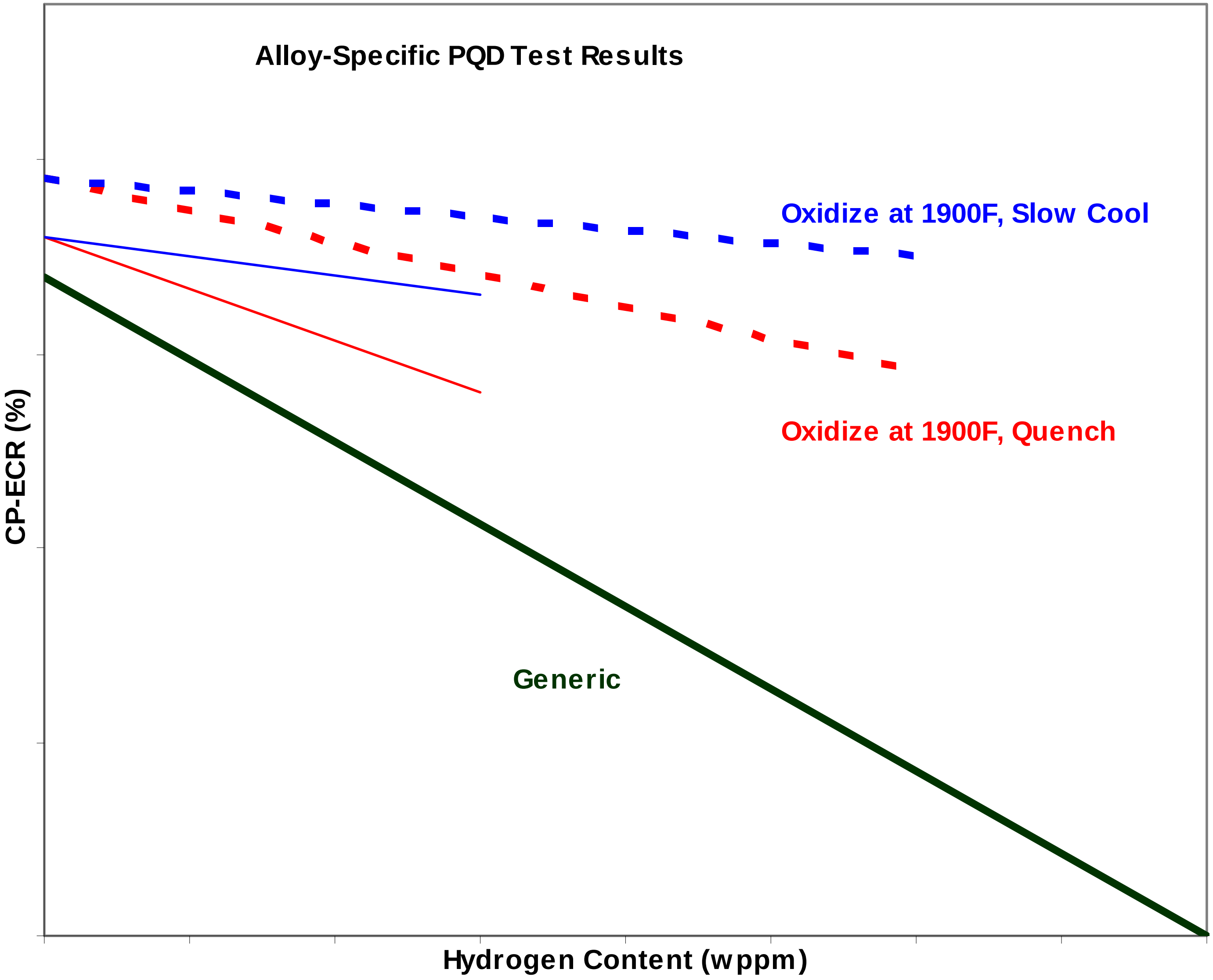
(initial hydrogen content converted to burnup)



Added Flexibility



Added Flexibility (cont.)



Path Forward

Ongoing Research Activities

1. Development and validation of a comprehensive, performance-based test procedure.
2. Additional PQD tests at intermediate hydrogen levels.
3. Additional breakaway tests to investigate whether the timing of breakaway oxidation is sensitive to variations in temperature profile or thermal cycling.

Advance Notice of Proposed Rulemaking

- ANPR process designed to enhance public participation during significant rulemaking campaigns. Benefits include:
  - Public response to rule concept and/or staff requests for additional information factored into the rulemaking proceeding and language of proposed rule language
  - Facilitates formal stakeholder interaction on the rulemaking while further research is acquired.

Staff White Paper Concerning Containment Overpressure Credits: Risk Considerations

Marty Stutzke, RES/DRA

ACRS Presentation
December 4, 2008

Staff Position

- The staff will continue to consider risk insights in its reviews of license amendment request (LARs) that contain requests for containment overpressure (COP) credits in accordance with its existing processes, which implement Commission-approved guidance.

Use of Risk Insights to Support Regulatory Decisionmaking

- NRR Office Instruction LIC-101 describes the staff's process for reviewing LARs.
- Risk-informed LARs
  - Guidance: RG 1.174 and SRP Section 19.2.
  - Risk insights provide one of the primary justifications for acceptability of the LAR.
  - All five key principles of risk-informed decisionmaking stated in RG 1.174 should be met.
  - Licensees voluntarily submit risk informed LARs.
- Non-risk-informed LARs
  - Guidance: SRP Section 19.2, Appendix D.
  - Risk insights may be used to determine whether or not a proposed plant change rebuts the presumption of adequate protection despite the fact that the proposed change meets currently specified regulatory requirements.
    - If one or more of the five key principles are not met, then a more complete assessment (deterministic and/or probabilistic) should be performed.
    - The fact that one or more of the five key principles is not met does not automatically imply a lack of adequate protection (i.e., the five key principles do <u>not</u> define "adequate protection").
  - Staff assumes the burden of demonstrating that the presumption of adequate protection is not supported.

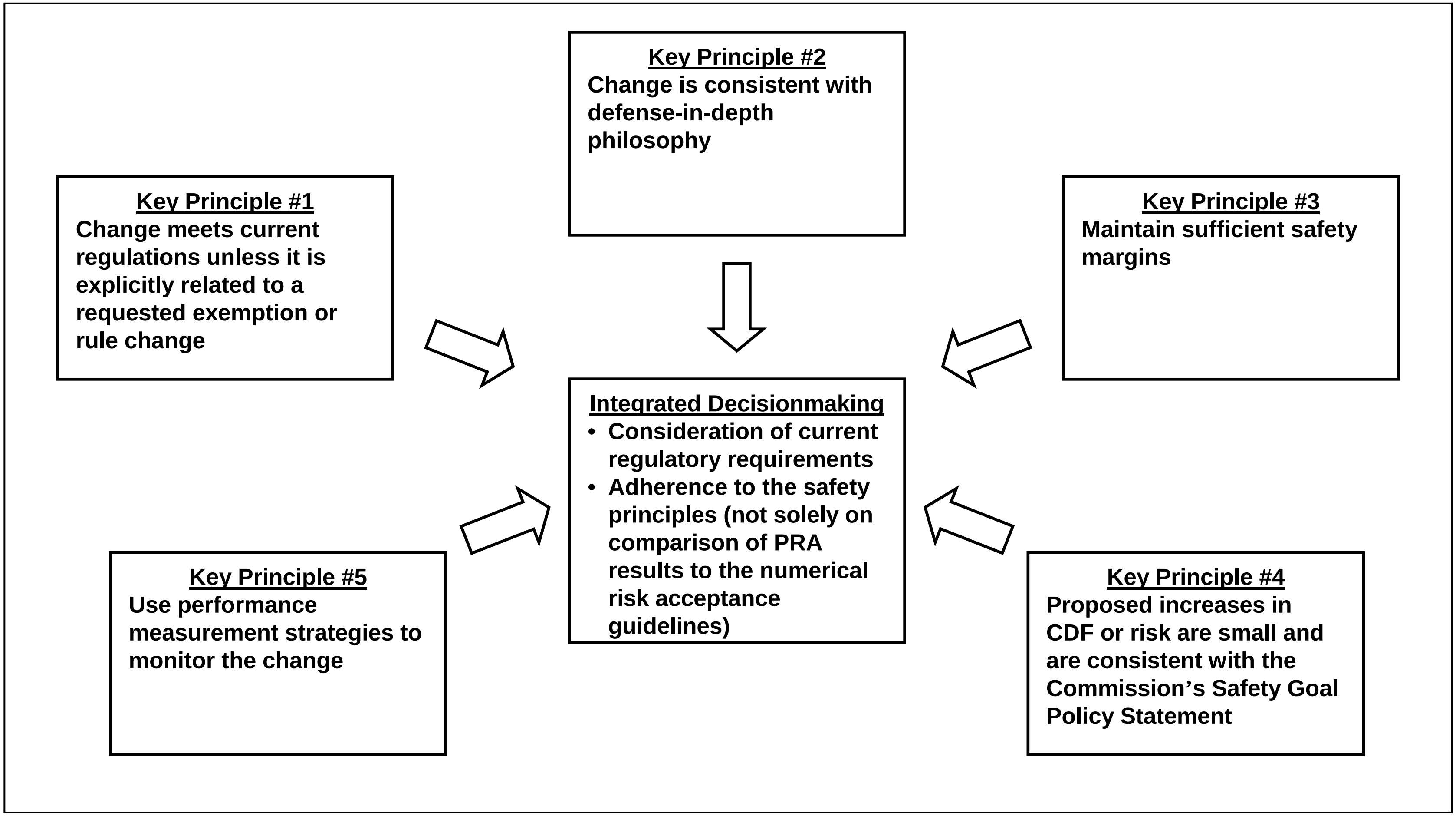
Invoking SRP Section 19.2, Appendix D

- SRP Section 19.2, Appendix D is invoked when the staff believes that a non-risk-informed LAR:
  - – Significantly changes allowed outage time, initiator probability, mitigation probability, recovery time, or operator action,
  - – Significantly changes functional requirements or redundancy,
  - – Significantly affects the basis for successful safety function, or
  - – Creates “special circumstances:”
    - Substantially increases the likelihood or consequences of accidents that are risk significant, but beyond the design and licensing basis of the plant,
    - Degrades multiple levels of defense or Reactor Oversight Process cornerstones,
    - Significantly reduces availability/reliability of systems, structures and components that are risk significant, but not required by regulations, or
    - Synergistic or cumulative effects that significantly impact risk.

Using SRP Section 19.2, Appendix D

- The numerical risk acceptance guidelines and safety principles in RG 1.174 are intended to provide a basis for finding that there is reasonable assurance of adequate protection.
  - The guidelines and safety principles serve as a point of reference for gauging risk impact, but are not legally binding requirements.
  - SRP Section 19.2, Appendix D emphasizes the need to differentiate between the concept of adequate protection and the numerical risk acceptance guidelines.
- The staff must notify the Commission whenever “special circumstances” are identified.
- The decision to reject a non-risk-informed LAR on the basis of risk will be made by the Director, NRR.

Five Key Principles of Risk-Informed Decisionmaking



Key Principle #2
Change is consistent with defense-in-depth philosophy

Key Principle #1
Change meets current regulations unless it is explicitly related to a requested exemption or rule change

Key Principle #3
Maintain sufficient safety margins

Integrated Decisionmaking
- Consideration of current regulatory requirements
- Adherence to the safety principles (not solely on comparison of PRA results to the numerical risk acceptance guidelines)

Key Principle #5
Use performance measurement strategies to monitor the change

Key Principle #4
Proposed increases in CDF or risk are small and are consistent with the Commission's Safety Goal Policy Statement

Application of the Five Key Principles to COP Credits

- Principle #1: Compliance with regulation
  - There is no regulation that prohibits use of a COP credit.
- Principle #2: Defense-in-depth
  - RG 1.174 focuses on understanding how a proposed change affects the physical barriers that provide defense-in-depth.
    - A COP credit reduces defense-in-depth because it introduces a dependency between the containment and fuel cladding barriers.
  - SRP Section 19.2 also discusses the need to consider programmatic elements that provide defense-in-depth.
    - In and of itself, a COP credit does not eliminate or alter any programmatic element (i.e., containment leakage testing) that provides defense-in-depth.
    - Licensees and staff should consider possible synergistic effects that may arise when various programmatic elements are modified (perhaps through a series of LARs).

Application of the Five Key Principles to COP Credits (Con't.)

- Principle #3: Safety margins
  - Discussed elsewhere in the staff's presentation.
- Principle #4: Small changes in risk
  - Current estimates indicate that the change in internal events CDF due to a COP credit is less than 10^{-6}/y.
  - The staff's white paper describes how PRA elements should be modified to reflect a COP credit.
  - RG 1.174 allows the use of qualitative risk evaluations (e.g., seismic margins analysis).
  - The final acceptability of a proposed COP credit is based on consideration of current regulatory requirements and adherence to the five key principles, and <u>not solely</u> on a comparison of quantitative PRA results to the numerical risk acceptance guidelines.
- Principle #5: Performance measurement
  - The staff's white paper lists many performance measurement strategies relevant to COP credits.

Backup Viewgraphs

Acronyms and Initialisms

COP	containment overpressure
LAR	license amendment request
NRR	The Office of Nuclear Reactor Regulation
RG	Regulatory Guide
SRP	Standard Review Plan (NUREG-0800)

The Evolution of SRP Section 19.2, Appendix D

- 8/25/1997, COMSAJ-97-008: Discussion of compliance and safety; staff has the responsibility to consider risk during review of LARs.
- 4/12/1998: Union Electric submitted LAR to electrosleeve SG tubes at Calloway (not risk-informed); staff concerned about behavior of electrosleeve material during severe accidents.
- 12/23/1998, SECY-98-300: Options to risk-inform 10 CFR 50; staff identified policy issue to get clarification of its authority to apply risk-informed decisionmaking in areas beyond those associated with licensee-initiated risk-informed LARs.
- 5/24/1999: Staff approved Calloway electrosleeve LAR.
- 6/8/1999, SRM on SECY-98-300: Commission agreed that additional guidance was needed.
- 10/12/1999, SECY-99-246: Transmitted interim guidance on applying risk-informed decisionmaking in LARs.
- 1/5/2000, SRM on SECY-99-246: Commission approved interim guidance.
- 3/28/2000, RIS 00-007: Advised licensees about interim guidance on the use of risk information by the staff during its reviews of LARs.

The Evolution of SRP Section 19.2, Appendix D (Con’t.)

- 4/10/2000: Draft appendix to SRP Chapter 19 published in the Federal Register.
- 5/11/2000: ACRS meeting on draft appendix to SRP Chapter 19.
- 5/16/2000: Public workshop on draft appendix to SRP Chapter 19.
- 5/30/2000: CRGR meeting on draft appendix to SRP Chapter 19.
- 9/26/2000: Staff forwarded the final appendix to SRP Chapter 19 to the Commission.
- 11/12/2000, COMSECY-00-0038: Commission approved final appendix to SRP, Chapter 19; directed the staff to notify the Commission of the first few LARs that create special circumstances.
- 1/18/2001, RIS 01-002: Advised licensees of final guidance on the use of risk information by the staff during its reviews of LARs.
- June 2007: Former SRP Chapter 19 redesignated as SRP Section 19.2; Appendix D retained without modification.

Use of Containment Accident Pressure in Determining Available NPSH of ECCS and Containment Heat Removal Pumps

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December 4, 2008

PURPOSE

- To discuss the NRC staff position on the use of containment accident pressure in determining the available NPSH of ECCS and containment heat removal pumps

- Staff position and discussion provided to ACRS in a memorandum to the ACRS Executive Director, dated November 4, 2008

TOPICS

- INTRODUCTION
- REGULATORY BACKGROUND
- REGULATORY BASIS
- TECHNICAL BASIS
- RISK CONSIDERATIONS
- FUTURE ACTIONS
- CONCLUSIONS

INTRODUCTION-1

- Changes to November 4, 2008 position paper:
  - Position paper states that RG 1.1 will be withdrawn. It will not. (Executive Summary and Page 11)
  - For non-EPU submittals, risk procedure will follow SRP 19.2 Appendix D (Executive Summary and Page 28)
  - Discussion of uncertainty in NPSHR will be revised (Page 4)

INTRODUCTION-2

- ECCS AND CONTAINMENT HEAT REMOVAL PUMPS IN BWRs AND PWRs ARE CENTRIFUGAL PUMPS
  - Capable of operation over a wide range of flow rates and pressures
  - Operation well understood
  - Used in wide variety of applications
  - Subject to cavitation

REGULATORY BACKGROUND-1

- Regulations allow use of containment accident pressure in determining the available NPSH of safety related pumps

REGULATORY BACKGROUND-2

- RG 1.1 November 1970
- RG 1.82 Rev 0
  - 50% Blockage
- RG 1.82 Rev 1 November 1985
  - Incorporates findings of USI A-43
  - Uniform coverage of sump screens by loca generated debris
  - RG 1.1 cited as guidance for use of containment accident pressure

REGULATORY BACKGROUND-3

- RG 1.82 Revision 2 May 1996
  - Incorporates guidance supporting NRC Bulletin 96-03
- GL 97-04 October 1997
- RG 1.82 Revsion 3 November 2003
  - Incorporates guidance supporting NRC Bulletin 2003-01
- DRAFT RG 1.82 Revision 4
  - Revises guidance on calculating available NPSH
  - September 20, 2005 ACRS letter recommended revisions and further restrictions on use of containment accident pressure prior to issuing

NRC POSITION

- The NRC allows use of containment accident pressure in determining available NPSH in the following cases:
  - Analyses using conservative assumptions have demonstrated that this pressure will be available for postulated design basis accidents
  - When examined from a broader perspective (i.e., beyond design basis accidents), an acceptable level of safety is maintained

NRC POSITION-2

- Duration of use of containment accident pressure is not risk significant.
- Significant contributors to loss of containment integrity occur at start of postulated accident:
  - Pre-existing leak
  - Failure of containment isolation
  - Possible exception for App R fire (associated circits). Examined during staff reviews.

NRC POSITION-3

- The magnitude of pressure needed is not risk significant.
- A calculation of peak LOCA containment pressure demonstrates that the pressure is less than the design pressure.
- Pressure at the time of peak sump or suppression pool temperature is much less than containment design pressure.

NPSH MARGIN

- Some authorities specify a margin between NPSHR and NPSHA of 30% or more
- Nuclear industry practice is NPSHA = NPSHR
- This is acceptable because:
  - LOCA pressure is conservatively calculated
  - Margin is important to ensure continuous long-term pump operation. Not one time operation for period of hours.
  - Tests have shown that damage rate is highest at some point between 3% and incipient cavitation. (Pump dependent.)

TECHNICAL BASIS

- Considerations for acceptability of using containment accident pressure:
  - High confidence in containment integrity
  - Conservative calculations
  - Pump design
  - Emergency operating procedures
  - Minimal impact on plant risk

CONTAINMENT-1

- RG 1.1: One rationale for not using containment accident pressure is the possibility of “impaired containment integrity”
  - Structural integrity test prior to licensing
  - 10 CFR 50.54(o) and Appendix J require leak testing of containment and individual penetrations
- 10 CFR 50.55a requires periodic inspections of the containment.
- TS control containment integrity.
- Stringent plant procedures.
- Good experience

CONTAINMENT-2

- Majority of plants using containment accident pressure to determine available NPSH are BWRs with Mark I containments
  - inerted
  - O_2 monitors
  - Drywell-wetwell ΔP restricted by technical specifications

CONTAINMENT-3

- 4 plants subatmospheric. 3 more operate as sub- atmospheric
- 4 PWRs with large dry containments

CONTAINMENT-4

- Other safety analyses assume containment integrity:
  - Containment integrity is assumed in calculating offsite dose (10 CFR 50.67 or 10 CFR Part 100)
  - Accident pressure is assumed in calculating peak cladding temperature (10 CFR Appendix K)

EQ CONSIDERATIONS

- SRP 3.11 covers all items of equipment important to safety (mechanical, electrical, I&C)

- SRP 3.11:
  - *For mechanical equipment located in a harsh environment, compliance with the environmental design provisions of GDC 4 are generally achieved by demonstrating that the nonmetallic parts/components are suitable for the postulated design basis environment conditions.*

  - *For mechanical equipment, the staff concentrates its review on materials that are sensitive to environmental effects (e.g., seals, gaskets, lubricants, fluids for hydraulic systems and diaphragms*

CONSERVATISM-1

- Calculation for LOCA underestimates containment pressure and overestimates suppression pool or sump temperature
- Calculations for ATWS, station blackout and Appendix R fire are realistic
  - some conservatism is typically present
- NRC staff November 4, 2008, white paper provides lists of typical conservative assumptions used in BWR and PWR LOCA calculations.

PUMP DESIGN-1

- All pumps of interest share certain characteristics with respect to cavitation:
  - robust construction
  - mechanical seals
  - stainless steel (cavitation-resistant) impellers
- ECCS pumps of later plants have lower required NPSH than those used in earlier plants

PUMP DESIGN-2

- NRC staff has approved pump operation in cavitation below the required NPSH
- Based on testing and subsequent inspection of pumps
- These tests of prototypical pumps in cavitation have not shown damage or more than very minor wear (scratches)

PUMP DESIGN-3

SUMMMARY OF NUCLEAR POWER PLANT SAFETY RELATED PUMP CAVITATION TESTING

PLANT	PUMP	COMMENTS
Browns Ferry	RHR	●Tests performed at 8000 and 10000 gpm ● Severe audible cavitation but acceptable motor vibration ● Tests terminated before "breakout" point (complete loss of head) ● Discharge head drop 10-12% ● Manufacturer's NPSHR curves may be reduced an additional 9 ft ● Operated for 10 minutes below manufacturer's recommended design NPSH conditions
Browns Ferry	RHR and Core Spray	● Pump vendor provided curves showing acceptable operation for limited times at up to 6% head loss ● Based on total operation time of 8000 hrs at various NPSHR values.
Dresden	Core Spray	● Witness and NPSH testing. Pump disassembled and examined. All parts in excellent condition. ● Cavitation tested 4000 to 6000 gpm. Time not specified. Pump disassembled and examined. No damage or wear. ● Pump again tested below previous cavitation point for one hour. No damage or wear. ● Pump cavitation tested again for one hour. Suction pressure lowered and tested further for 30 minutes. Pump again disassembled and examined. No damage or wear. Several scratches.
Vermont Yankee		● Pump vendor provided curves showing acceptable operation for limited times at up to 6% head loss ● Based on total operation time of 8000 hrs at various NPSHR values.
Monticello	Core Spray	Cavitation test performed by pump vendor. Pump went through "extensive cavitation" for several hours "without visible damage to the impeller."
Beaver Valley (North Anna Unit 2 pump)	Recirculation Spray	Closed loop test. NPSHA lowered by water temperature increase and tank level decrease Initial NPSHA = 15.1 ft. NPSHA lowered to 5 ft (well into the breakdown region) for ½ hour. After testing pump Total Dynamic Head/Capacity curve regenerated. No degradation noted.
Crystal River	Building Spray	Pump vendor provided justification for a required NPSH based on a 5% head drop

OPERATIONAL CONSIDERATIONS-1

- BWR EOPs consider containment pressure in assessing adequate available NPSH
- BWR NPSH analyses consider operation of containment spray for the duration of the event

OPERATIONAL CONSIDERATIONS-3

- Operator indications of cavitation (from control room)
  - Erratic or decreasing pump motor current
  - Erratic flow or flow less than expected
  - Frequent adjustments to ECCS pump discharge valves to maintain constant flow rate (BWRs)
- Operator response to cavitation
  - throttle pump
  - remove pump from service
  - consider other water sources

EFFECT OF THROTTLING

Dresden 2/3 calculation DRE97-0002 Rev 0 Attachment A

RHR/CS	RHR Pump Flow/Pump (gpm)	Suction Loss (ft)	NPSHR (ft)	NPSH margin (ft)
4/2	5000	10.7	30	-11.1
4/2	3750	6.5	25.5	-0.7
4/2	2500	3.4	25	4.3

FUTURE ACTIONS

- Revise RG 1.82 Revision 3
  - Clarify and add more detail to NPSH discussion
  - Revise positions
  - Remove material not relevant to current status of the issue (e.g., sump design descriptions)
  - Update references
  - Revise RG 1.1 to state that RG 1.82 provides the current guidance.
- Revise white paper
- Make white paper publicly available.

CONCLUSIONS

- High confidence in containment integrity
- Prototypical pumps have been cavitation tested for periods up to several hours with no damage
- Need for credit for containment accident pressure for BWRs is limited to older plants
- Where examined, the risk of using containment accident pressure in determining available NPSH is negligible
- For some plants, reliance on containment accident pressure is a result of conservative analysis

BACKUP SLIDES

STATISTICAL APPROACH

- A statistical estimate of the uncertainty in the pressure needed for adequate NPSH is added to a realistic value
- BWROG has submitted NEDC-33347 for review and approval.
- Approach used in several other areas of reactor safety analysis:
  - Realistic LOCA
  - Departure from Nucleate Boiling Ratio (DNBR)
  - BWR Anticipated operational Occurrences (AOO’s)

PENETRATION SEALS

- “Both Viton and EPR/EPDM O-rings appear undamaged when exposed directly to a steam environment with temperatures up to about 600 F at a pressure of 155 psia for 4 to 6 hours…
- “Silicon rubber O-rings appear undamaged up to 500 F at a pressure of 155 psia when exposed directly in a steam environment for about 4 hours…”

8th SMIRT Conference 1985

“Integrity of Containment Penetrations under Severe Accident Conditions,” C.V. Subramanian