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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

November 2, 2007

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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547TH MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARD

(ACRS)

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FRIDAY

NOVEMBER 2, 2007

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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:

WILLIAM SHACK, Chairman

MARIO V. BONACA, Vice Chairman

DANA A. POWERS, Member

JOHN D. SIEBER, Member

SANJOY BANERJEE, Member

DENNIS BLEY, Member

J. SAM ARMIJO, Member

SAID ABDEL-KHALIK, Member

COMMITTEE MEMBERS (CONT.)

OTTO L. MAYNARD, Member

JOHN W. STETKAR, Member

MICHAEL CORRADINI, Member

NRC STAFF PRESENT:

STEVE ALEXANDER

AMY CUBBAGE

DON DUBE

BOB DAVIS

KIM GRUSS

ALSO PRESENT:

JIM KINSEY

ALAN BEARD

RICK WACKOWIAK

JERRY DEAVER

BRIAN FREW

AGENDA

P R O C E E D I N G S

(8:30:51 a.m.)

CHAIRMAN SHACK: The meeting will now come to order. This is the second day of the 547th Meeting of the Advisory Committee on Reactor Safeguards. During today's meeting, the Committee will consider the following; one, selected chapters of the SER associated with the ESBWR design certification; two, future ACRS activities and report of the Planning and Procedures Subcommittee; three, reconciliation of ACRS comments and recommendations; four, the draft ACRS report on the NRC Safety Research Program; and, five, preparation of ACRS reports.

This meeting is being conducted in accordance with the provisions of the Federal Advisory Committee Act. Mr. Kenny Santos 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 sessions. A transcript of a portion 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.

I'd like to remind the members that we are scheduled to interview a candidate during lunch time today, so don't disappear too far.

With that, our first topic today is some review of chapters of the SER associated with ESBWR design certification, and Mike Corradini will be leading us through that.

MEMBER CORRADINI: Thank you, Mr. Chairman. So many of the members were at our Subcommittee meeting, but let me try to summarize where we are. So General Electric-Hitachi submitted their ESBWR design certification back in August of 2005, and based on staff requests, GE submitted additional materials, and the staff formally accepted the application in December of '05.

The staff then issued a request for additional information, and based on that and the original application, GEH responded to these RAIs, and the staff prepared a preliminary SER with open items. It's a bit different in the sense that the staff has asked us, and we felt it was appropriate, to start looking at the SER in chapter-by-chapter format. So what we did, and if you remember, on October 2nd and 3rd, had first a Subcommittee meeting, which kind of looked at the overview of the ESBWR, where GEH and the staff came in and did, I think, a very nice job of explaining to all of the rookies, at least, the new system, how it functions, how all the system components fit together in terms of interactions. Then on the 3rd, we had a meeting where we looked at three of the chapters of the SER, Chapters 2, 8, and 17, and then, subsequently, Wednesday we had another meeting where we looked at three other chapters, 5, 11 and 12. And since you don't know the numbers, let me remind you that encompasses topics of site characteristics, the power conversion system, electrical power, both RAD protection and radioactive waste management, and quality assurance. That was a test for me to remember all these things.

But, in any case, those six topics we looked at. We reviewed what the staff had done relative to RAIs, I'm sorry, in terms of open items, and discussed with them, and with GEH, our concerns, questions, comments, et cetera. And, so, the purpose of today is to hear a summary from that discussion, and then our intent is to probably write a letter discussing what we think about relative to those six chapters, and any comments we can give to the staff relative to things occurring there, or how they might fit in with further things we'll see. And, so, I'll turn it over to Amy Cubbage.

MS. CUBBAGE: Sure. Thank you. Amy Cubbage, I'm the Lead Project Manager for ESBWR. I'd like to thank the Committee for allowing us to come with a chapter-by-chapter basis on the safety evaluation. At this time, we've asked over 3,000 RAIs, and 2,300 of them have now been considered resolved. And the additional 800 open items are identified in the safety evaluation reports that have been provided to the Committee.

Our approach is to engage the Committee on this chapter-by-chapter basis to obtain your early feedback on any issues that you have. Our goal for the meeting is two-fold. We are requesting a letter from the Committee, and in addition to getting any feedback you have, so we can address them as part of the review.

When we come back to the Committee in 2009 with the complete SER, we're going to be focusing on the resolution of the open issues, and any changes that have occurred as a result of GE's changes to the DCD. With that, I'd like to turn it over to GE, Jim Kinsey.

MR. KINSEY: Good morning. This is Jim

Kinsey with GE-Hitachi. I was just going to give a couple of introductory remarks, and cover the first slide or two in our session, and then I'll turn things over to our presenters, who are in the front of the room.

I'd also like to thank the Committee in this format of covering individual chapters earlier in the process, I guess, than normal, and again on a chapter-basis, so that we can work through the specific issues, and questions, and concerns you may have.

As Dr. Corradini mentioned, we've covered six chapters at this point. Our intention this morning, from GE-Hitachi, is to just give you a very brief overview of those six chapters. Based on some of the questions and comments that we had in the last session, we have a couple of additional slides in the package this morning covering Chapter 5, which is associated with the reactor coolant system. Then we've put that group at the back of the presentation, so we can go through that again in a little bit more detail.

We've structured the presentation, again, to be a very brief overview. We'll move through that probably relatively quickly, and then turn things over to the NRC staff to continue on with the presentation.

I'll go on to the next slide. Again, just as a summary or a review, we've previously discussed Chapters 2, 8, and 17, and the Subcommittee Activity on October 3rd, and then moved on to Chapters 5, 11, and 12 on October 25th. And I'll turn things over to the team in the front of the room to step through each of those chapters briefly. Thank you.

MR. BEARD: Okay. Good morning. My name is Alan Beard. I think many of you know me from previous presentations. I'm going to cover all but Chapter 5 at a very high level. If you have questions, please feel free to ask them, and I'll try to respond to those as best as possible, but we're going to focus on the issues that came up during the Subcommittee review.

So, on Chapter 2, just to refresh your memory, this deals with the bounding site characteristics, and things like meteorology, hydrology, geology, seismology, geotechnical parameters, and any potential nearby hazards. Let me say that we took some lessons learned in this area. The EPRI URD has established what they felt was a bounding site set of conditions. As you saw in the Early Site Permits coming in, and we identified some of those Early Site Permits that were coming in with values that were not bounded by the URD values, so we did increase some of our values. Examples of that are some of our wet bulb temperatures are in excess of the URD temperatures, as well as some of our wind speeds. Our tornado wind speed is about 10 percent greater, and our maximum wind speeds for hurricane condition are also 10, 15 percent greater than they would have been under the URD.

An applicant referencing the ESBWR DCD for a COL will have to demonstrate to the staff that the site conditions at that site are bounded by the ones that we used in the basic design, and let me leave it at that. Are there any questions on anything we're doing in this particular area? I think it's pretty consistent with what you've seen in previous designs. The only difference is we have improved some of our design parameters.

Okay. Next slide.

CHAIRMAN SHACK: You do -- you are going to design, though, with a different kind of seismic spectra than the Standard Reg Guide 160.

MR. BEARD: Right. Our seismic spectra is the Reg Guide 160, and then it has an additional overlay extending that for about 100 hertz. And that is in DCD Chapter 2. There is a figure for that.

CHAIRMAN SHACK: So that will be part of the certification.

MR. BEARD: That is part of the certification, yes.

Okay. Chapter 8, the electrical system includes discussion of the off-site power distribution to the extent that we can do that as part of the certified design, so that really goes just out through our main transformers, and then the switch yard is part of the site-specific design. Then we look at the on-site power distribution. With our design, we have power generation process called Plant Investment Protection Busses. Those are the busses we're using for our Defense In-Depth Systems, primarily, and also, like it says, plant protection, power things like new boil pumps and all that, in case we do have loss of off-site power. We do preserve the capital investment in the plant.

And the real focus in most of the chapters are safety-related system. Because we are a passive plant, we are relying upon massive quantities of batteries to provide stored energy. Most of that DC energy is converted back to AC, when we use it in safety-related applications, very little of it actually goes out as DC power for the safety-related applications.

And then just as a note, the detailed station blackout analysis is actually discussed in Chapter 15. I'll give you a preview of that. We do have a 72-hour capability without any kind of on-site or off-site AC power. However, when we do the analysis following the guidance in the NUMARC, I forget the document number, but the reference that has been approved by the NRC, we expect that the maximum duration of a station blackout would be on the order of eight hours. So we have a lot more capability than what we actually need. And that was the only slide I had for Chapter 8, so again I will pause, and see if there are any questions.

MEMBER MAYNARD: Well, as I recall in our Subcommittee, there was some discussion about the battery life, whether the battery life was 72 hours or whether it was 36.

MR. BEARD: Okay. The answer to that is we have four divisions of batteries, each one of those divisions has a 72-hour capability. Within the division, we actually have two different batteries, each of which is nominally a 36-hour capability, but we treat the division as a division. We don't treat those individually.

MEMBER STETKAR: To follow-up on that, because I raised part of the question during Subcommittee. I'm reading from DCD Rev. 4, which is dated September 2007, and it says, "Figure 8.1.3 shows the overall 250 volt DC system provided for safety-related divisions 1, 2, 3, and 4. Divisions 1, 2, 3, and 4 consist of two separate batteries in each division. Each battery supplies power to its safety-related inverter for at least 72 hours, following a design-basis accident." Each battery.

MR. BEARD: Okay. We need to -

MEMBER STETKAR: Seventy-two hours.

MR. BEARD: The intent is we have two batteries with nominal 36-hour capability.

MEMBER STETKAR: There still seems to be confusion about that.

MR. BEARD: I believe that's just -

MR. KINSEY: This is Jim Kinsey from GE-Hitachi. I think the intention of those words was to indicate that both portions of that division's battery are in service for the 72-hour period, but the way it's written gives you the impression that they each have a 72-hour capacity to carry the division.

MEMBER STETKAR: It certainly gives me

that impression.

MR. KINSEY: We need to take a look at the words. That was another -

MR. BEARD: We will take the action to go back and revise it so that it's clear that we're talking about -

MEMBER ARMIJO: I don't know anything about batteries, but is this an unusual application? Is there any experience with this big a set of batteries in nuclear plants?

MEMBER SIEBER: Submarine batteries. Regular power plant batteries don't run like this.

MR. BEARD: For the active plants, yes, we definitely have significant larger -

MEMBER MAYNARD: But large banks of batteries are pretty common.

MEMBER ARMIJO: So there's nothing -- it's not an order of magnitude-type application to increase the -

MR. BEARD: No, we're just doing parallel - add on to it. We just increase the size of the bus bars, and add in parallel that to increase your capacity.

MEMBER SIEBER: They've been making industrial batteries for 60, 70 years.

MEMBER MAYNARD: I think it's an important issue of wording, of how it's worded in here, than it is -- I think you're meeting the requirements here in the 72 hours.

MR. BEARD: Yes.

MEMBER MAYNARD: I think I understand. If everything is working fine, each one of those batteries is -- they're 36-hour batteries, but they're only go to be at half load, so they'll operate for 72 hours. But the wording would kind of give you the impression that each individual battery is a 72-hour battery.

MR. BEARD: I agree with your interpretation, and we will take the action and go back and clarify that we're talking about it at a divisional level, not individual battery packs within divisions.

MEMBER SIEBER: It would be helpful if you said whether you had to switch over, or the batteries are always in parallel.

MR. BEARD: The batteries are always in parallel. There is a static transfer switch that automatically -- normally, we're powering through an inverter, and then back through a rectifier to feed the normal safety-related power out to the loads, the batteries are certainly on standby, and there is a transfer switch that automatically, when we lose that feed, will allow the batteries to feed them.

MEMBER SIEBER: There ought to be a way to say that in one sentence.

MR. BEARD: We do have the action item. We'll go back and re-look at that, and make sure that we do clarify that we're talking about it at a divisional level, and not sub-trains within the division. Any other questions or comments?

Chapter 11, radioactive waste management looks at the source term calculations, the fission activation products, and after that looks at liquid waste management, gaseous waste management, and process and effluent monitoring and sampling.

In the area of gaseous waste management, very typical of what conventional BWRs use, interlock gas treatment system, no big surprises there. And the liquid waste management system, we are following industry guidance, and expectations that we'd go to a modular-type approach. What I mean by that is we have designed a building. That building has significant volumes of tanks and pumps to accumulate the local rad waste, but the actual processing of those streams will be done using various skids that can be brought in on the grade elevations, and temporarily plumbed into the system to process the fluid streams, as necessary, depending on what you're trying to accomplish at that one particular time. And that, again, at a high level I think is all I really needed to talk to, so, again, I will stop and see if there's any questions or comments.

MEMBER SIEBER: So once you're through using the skid, you have a contaminated piece of equipment you've got to get rid of. Right?

MR. BEARD: Actually, the industry thinking is that that is a subcontractor service, and -

MEMBER SIEBER: No, couldn't -

(Laughter.)

MR. BEARD: It's somebody else's headache. That's exactly right.

MEMBER SIEBER: You end up buying trucks when you do it that way.

MEMBER MAYNARD: I don't think the intent is just be switching these in and out all the time. I think it's taking a look, in reality right now, the plants -- most of them are not using the original designed systems. They've replaced them with a skid, but the original design didn't really allow for that.

MR. BEARD: Right. Probably half the plants out there that have rad waste buildings that have been abandoned, and brand new ones built because they had to abandon all the equipment in place.

MEMBER CORRADINI: So you're still in the process, just to make sure I understand, as we left it, you're still in the process of specifying in more detail the specs on these, I won't call them mobile, but these units.

MR. BEARD: No, we're not going to specify those at this time. We know that there are units out there capable of the flow rates that we need to process.

MEMBER CORRADINI: But when I say spec it, in other words, the envelope that you require to fit into the system; that is, it's got to meet certain criteria.

MR. BEARD: Yes. Yes. Any other questions?

MS. CUBBAGE: It doesn't sound consistent with what the staff had been talking with GE about. I believe that GE is intending that in a future DCD rev, those mobile systems will be part of the certified design, rather than conceptual.

MEMBER ARMIJO: That's what I thought I

heard at the Subcommittee meeting, also.

MR. KINSEY: To embellish on Alan's answer, I guess maybe I heard what he said in a different way, but it is our intention to remove the conceptual design language from the DCD. We're embellishing, or adding to the description of the design, and it's our intention to add some additional figures reflecting those designs. And you'll see that in DCD Rev. 5.

MEMBER CORRADINI: So let me ask the question differently. I don't want to dwell too much on it, but I want to make sure. So in my simple mind, you're saying there -- at least the way I understood it in the Subcommittee, there are certain things that this system must do, and in the Subcommittee I heard that, let's pick examples, I will decide to pick a Chevrolet version of this, as essentially an example of what would satisfy the requirements. Am I off base here? That is, I got the impression you were going to specify enough that one would know there are available units that can meet the requirements that you specify.

MR. KINSEY: Our intention is to provide an adequate description in the DCD of what's required. There will be a graphical depiction and some simplified flow diagrams there, and we have confirmed in the industry that there are currently systems available that can satisfy the depiction or the requirements that we're putting in the DCD, recognizing that there may be different, or more improved systems available at points in the future that could also be -

MEMBER CORRADINI: They could fit that, but could actually have better performance.

MR. BEARD: Right.

MEMBER CORRADINI: Okay.

MR. BEARD: Exactly.

MEMBER CORRADINI: Is that -

MS. CUBBAGE: That's consistent with what the staff was expecting.

MR. BEARD: Did you have a comment?

MEMBER CORRADINI: No. Go ahead. Sorry.

MR. BEARD: Next slide then, please.

MEMBER CORRADINI: I think you hadn't talked yet about rad protection, did you?

MR. BEARD: Oh, I'm sorry. Chapter 12, Radiation Protection, provides an assessment of the various radiation sources within the power block, how we're going to protect the workers and the public from those radiation sources, provides a dose assessment of what the operational personnel we would expect would receive in occupational exposure on a yearly basis, talks somewhat about the health physics requirements, and then also how we're going about minimization of rad waste, and generation of any radioactive waste itself. Again, some of these are the easier chapters. Five is one you're going to have all the questions, so we'll get to that as quickly as possible. I will stop and pause again with my standard phrase of any questions or comments?

MEMBER SIEBER: Gee, I didn't have the opportunity to go to the Subcommittee meeting, but my experience in the area of ALARA is that it's enhanced if there are lots of work platforms, and pre-built structures. And I understand from reading the DCD that you plan on some of those, but I'm not sure the extent to which you would pre-stage and pre-construct those types of things for ALARA purposes.

MR. BEARD: For the ABWR, we have what we call a project design manual, which laid forth the requirements on where we're going to provide permanent platforms, and staging, and ladders, and hoist points, and clearance requirements around particular equipment and all that. We will be updating that project design manual and folding the known ESBWR as we detail the design to place those requirements on exactly where do we want to provide those permanent types of equipment.

MEMBER SIEBER: Okay. Those structures would be in GE's scope, and not the -

MR. BEARD: That's correct. They will be part of the standard design.

MEMBER SIEBER: Okay.

MR. BEARD: I mean, if there's a platform in one plant, there'll be a platform in the same plant when you go to it and look at -

MEMBER SIEBER: Well, if there isn't, should be, it won't be in any plant.

MR. BEARD: Correct. Now we go to Chapter 17. Okay. Chapter 17 is just a discussion of the Quality Assurance Program, talks at a high level of GE-Hitachi Quality Assurance Program, describes how we do control of our work processes, as well as qualifying our suppliers quality programs. And then it talks about what the Quality Assurance requirements we actually have as the base designer. And then the last portion in there, and I believe it's Section 17.4, talks about the Design Reliability Assurance Program, D-RAP is just to insure that as we detail the design, that the assumptions we made within the PRA are preserved as we detail the design. And if they're not preserved, we need to reconcile them.

MEMBER ABDEL-KHALIK: Where would something like surveillance programs and periodic replacement of things, like squib valves and so on, be covered?

MR. BEARD: Rick, do you have anything on that?

MR. WACKOWIAK: Good question. I don't believe that that information is covered in the DCD, in the scope of the DCD. We would be -- one of the things that we need to do is understand exactly which components it is that we'll be specifying for purchase and purchasing, and the periodic maintenance on components really can't be specified until we know what the components are.

MEMBER MAYNARD: Part of the surveillances will be defined as part of the tech specs in Chapter 16.

MR. BEARD: Well, the operational surveillance -

MEMBER CORRADINI: But some of these would be held off until the COL, I would assume, or am I incorrect in assuming that?

MR. BEARD: Well, some of them may even go beyond the COL, depending as we get final provider - let's talk the squib valves as an example - if they tell us they have a 12-year qualified life, we've already explained, I think that we expect to set up a rotational replacement over that 12-year, so that depending on what kind of outage cycle the utility chooses to do, they would be replacing anywhere from 15 to 25 percent of the squibs each outage, so that you're never going in and doing a wholesale replacement, in case you had a problem with a batch that might lead to a common cause failure postulation.

MEMBER ABDEL-KHALIK: That's really the reason for my question. I want to know where that information is going to be documented.

MR. WACKOWIAK: Let me try to answer the question this way. The D-RAP specifies a list of components that require monitoring, and maintaining a certain level of reliability. In the D-RAP program, we specify which components those are. The D-RAP then folds into an operational program like the maintenance rule, where we assure that the reliability levels that are assumed in the design are preserved or are better. So when you procure a specific type of component, the requirements on that component for maintenance, rotation of consumables or degradables, would be addressed at that time to say okay, if you're going to buy this particular component with this kind of life, you have to replace it on this type of frequency. If you're going to buy this particular component that has this operating history, or failure history, you would have to maintain it on a certain frequency in order to meet the reliability targets that were assumed during the design for the components that are in D-RAP. So what D-RAP is telling you is which things you need to worry about, the operational program later tells you how you actually implement that.

MEMBER SIEBER: But the fact is that you don't have replacements for every component in the plant. And that sometimes things fail that aren't part of your preventative maintenance program, and so you have to go and procure a replacement for it, and that's where engineering assurance and the quality assurance aspects, and a bounding description of what the component is supposed to do, which is in the DCD, that's how they're used to get a replacement.

MEMBER MAYNARD: I think most of the sites in question would probably be in the COL stage, because the licensee is going to have Quality Assurance Program requirements where he's going to have similar to what the Design Reliability Assurance Program is on the Operational Assurance Program that the procurement and the ongoing activities keep it there, so I think most of that would be in the COL, as part of that.

MEMBER SIEBER: That's going to evolve throughout the life of the plant, as the application of the Maintenance Rule shows that some component has a shorter life than the plant life time, a prudent operator who has arranged for replacements, or refurbishment, or whatever is necessary to keep that particular piece of equipment in service.

MEMBER MAYNARD: I think the important part for this stage is how this information going to be documented and transferred to the COL, how is that transition.

MR. WACKOWIAK: And as we said in the past, that is documented in one of our design specifications that is ongoing right now. As we said with the staff before, our expert panel, which is the final step in establishing the initial D-RAP set of components is scheduled for January, and the output of that will be in a GE design specification that is part of the standard design.

MEMBER CORRADINI: Which was the open item here. Right?

MR. WACKOWIAK: It was the open item, and closure of that open item is completion of that task.

MR. ALEXANDER: Excuse me. I wonder if I could add something. This is Steve Alexander from NRR. And the revisions that we've seen of the DCD do explain the reliability assurance process and the operations phase, and it includes not only maintenance rule, but very importantly, Quality Assurance, and Maintenance and Surveillance programs. And that's specified in the DCD, it's specified in our SER. It was called for in SECY 95-0132. All four of those programs, maintenance and surveillance, quality assurance, and maintenance rule to look at the effectiveness of that maintenance, and so that's a comprehensive set of operational programs to implement reliability assurance in the operations phase. If that helps.

MEMBER ABDEL-KHALIK: Just as long as that information is not lost. I mean, we've heard a lot of information, a lot of anecdotal, off-the-cuff response to many of these questions, and I want to make sure that that information is documented. Thank you.

MR. BEARD: Any other questions or comments? If not, I'm going to turn it over to Jerry Deaver and Brian Frew to talk to Chapter 5.

MR. DEAVER: Okay. I'll be talking about Chapter 5. The scope of Chapter 5 is to cover the reactor coolant system, which includes all the systems that bring effluent in or out of the core region. And also defined within Chapter 5 is what we call the reactor coolant pressure boundary, which is the same as typically in the past. It includes the high pressure piping and components in the containment, and includes the isolation valves, which include the isolation valves that are housed just outside the containment. And it includes the pressure relief valves that are part of the nuclear boiler system.

Also included in Chapter 5 is a description of the reactor pressure vessel. And on this slide, I've got several of the key changes that have been made in ESBWR. And one of the key features, safety features is the fact that there are no major vessel penetrations in the core region, or below. All we have are the typical bottom head penetrations, to include the control rod drive, and the in-core penetrations, and drain lines.

Another feature that has not been used in U.S. BWRs is use of large ring forgings in the past. We use two forgings for the closure flanges at the top of the vessel, but then we include four ring forgings and a disk for the bottom head, so all the materials from the vessel support region, which is just above core down to the bottom head uses large ring forgings. And we minimize welds by doing that, and we arrive at materials that are a better quality than disk plate materials.

Configuration-wise, the core region is two foot shorter within the BWR, ESBWR, and overall the vessel is taller primarily because of the chimney component, and the internal switch facilitates the natural circulation function of the ESBWR. The vessel is approximately 6-1/2 meters taller than ABWR or prior reactors, but the diameter is the same as the ABWR, 7.1 meters.

We have a little different venting system, which is a fairly minor change. The main steam line flow restrictors, this is a change that was made in ABWR. The main steam line nozzle itself contains the Venturi, that is the limiting feature for flow in the main steam line, which restricts the amount of break or discharge in a break scenario.

MEMBER ARMIJO: Is that a separate component that fits inside the nozzle, or is that -

MR. DEAVER: It's an integral part of the nozzle.

MEMBER ARMIJO: So is the nozzle forging shaped to provide that -

MR. DEAVER: Exactly. And then we have a little different support system to allow a sliding support as opposed to a rigid skirt system.

Okay. Next is a drawing that shows the overall configuration. Here you see that the vessel support area is above core, this being the core region itself. All of the basic components within the vessel are the same, basically, except for the chimney area, where we have an outer cylinder and partitions to direct the flow up above the core region. All other components, the separators, steam dryers, top guide core plate, the shroud are fundamentally the sam

DR. BANERJEE: How large are these partitions in the chimney?

MR. DEAVER: They're two feet by two feet. They encompass 16 fuel assemblies. So they're basically two foot by two foot, and 6-1/2 meters tall.

MEMBER ABDEL-KHALIK: When fully loaded, what is the elevation of the center of gravity of the vessel?

MR. DEAVER: I'm not sure I know exactly where -- it's got to be up probably a little bit below the actual physical center because of the fuel and the lower components.

MEMBER ABDEL-KHALIK: But it is above the supports. Is that correct?

MR. DEAVER: Yes, it is. Yes, I think in this design, the support header handles the actual movement of the vessel, because it's -- before we supported the vessel at the bottom, so it was more of a cantilever design, where you had to take a lot more seismic action on a cantilever basis.

MR. BEARD: And there are horizontal supports at the top, Jerry, if you want to point to them, as well.

MR. DEAVER: Yes. We still have the -

MR. BEARD: To provide lateral.

MR. DEAVER: -- lateral supports, vessel stabilizers in the same region that we normally have

DR. BANERJEE: How thick are the walls in the chimney region?

MR. DEAVER: The partitions or the -

DR. BANERJEE: The partitions.

MR. DEAVER: Nine millimeters. The actual internals will be covered in another chapter, which I believe is Chapter 4. This primarily includes - this chapter is the definition of the vessel, but not necessarily the internals. Okay?

MEMBER ARMIJO: Is this basically the same dryer design and location of steam lines, as in other

BWRs?

MR. DEAVER: Yes. The relative geometry in this region, the banks, and the steam nozzles are all very typical. We haven't made any changes in this area. Actually, what we're attempting to do right now is we're trying to -- because ABWR, the experience was very good as far as low vibrations and acoustic loads, we're trying to simulate as close as possible the ABWR configuration.

MEMBER CORRADINI: In the upper dome region.

MR. DEAVER: In the upper dome region, right.

DR. BANERJEE: And the velocities through the steam dryers and the lines are about the same?

MR. KINSEY: This is Jim Kinsey from GE-Hitachi. We appreciate the questions, but I guess we would prefer to cover questions associated with the reactor internals when we get to that.

MEMBER CORRADINI: That's Chapter 4. Is that correct?

MR. KINSEY: Chapters 3 and 4.

MS. CUBBAGE: Right. The steam dryer issues are specifically in Chapter 3.

MR. KINSEY: Right.

MS. CUBBAGE: And we have some ongoing discussions now with GE on the steam dryer issues reflecting the operating fleet issues.

MEMBER CORRADINI: I think if we ask too much, we're going to - although, these are highly expert individuals - they will run somewhere to call to get details, and we can catch them, I think, on the subsequent chapters.

MR. KINSEY: That was my suggestion, is we cover those detailed questions in that chapter, if that's all right.

MEMBER CORRADINI: January, Sanjoy, January

DR. BANERJEE: Mid-January.

MEMBER CORRADINI: Mid-January, Sanjoy.

MR. DEAVER: I'll cover two of the systems that are within the reactor coolant pressure boundary, one is the nuclear boiler system. This is your classical feedwater and main steam system. The steam lines have the SRVs and Svs located on the lines, similar to past reactors, and feedwater is essentially the same with six inlets into the vessel. The major change in this system are the DPV valves, which are located on the IC lines. We have eight DPVs that operate separately off the iso-condenser system.

Okay. The next slide shows the configuration of the steam lines and the relief valves and safety relief valves, as well as the DPVs, with a mixture of valves, SRVs and Svs, in each of the lines, so we have five in the large steam line, in the longer steam lines, and four in the short lines. The SRVs are the ones that could be manually operated, or just pressure actuated; whereas, the SVs are only pressure actuated. DPVs only actuate on a signal associated with containment isolation.

Okay. The next system will be the isolation condenser system. This is a system that was used on early BWRs, but is now being used in ESBWR, primarily to prevent actuation of safety relief valves. It's a means to discourage heat and energy after a containment closure. This system, the primary feature to the iso-condenser switch are located just outside of containment, and we have the steam lining that comes up into the condensers, and then we have the return line, which brings water back into the vessel. The key feature is the two parallel valves in the system, which always remain closed, unless the system is to be actuated. With these valves closed, then we get a backup of condensate in the system up to the condenser themselves, and this is the only part of the system that is closed. Okay. Any questions in iso system?

Okay. Also, in Chapter 5 we cover materials. We basically use common materials for valves -

MEMBER ABDEL-KHALIK: I think the question was brought up at the Subcommittee meeting, as to the effect of loss of nitrogen on the operation of the iso-condenser system, where that nitrogen operated isolation valve would actually fail open. And whether the sudden sort of entry of a large volume of cold water into the vessel during operation. I guess that particular transient has been analyzed and shown to be non-limiting?

MR. DEAVER: Right. Yes. That definitely has been analyzed. Yes, we have inadvertent operation of this system as part of one of the events that's analyzed in Chapter 15.

MEMBER ABDEL-KHALIK: Thank you.

MEMBER STETKAR: Does the Chapter 15 -- I haven't seen the Chapter 15 analysis. Does that analysis account for inadvertent operation of a single isolation condenser, or all four?

MR. DEAVER: Do you remember, Alan?

MR. BEARD: I'll look it up.

MEMBER STETKAR: Because loss of -

MR. BEARD: We'll check on that.

MEMBER STETKAR: Common loss of nitrogen would inject all four.

MR. BEARD: But there are accumulators designated to each of those valves.

MEMBER STETKAR: For how many minutes, considering normal leakage? That's too much detail, we'll keep going.

MEMBER ABDEL-KHALIK: But if the four of them were to open on loss of nitrogen, that's a very severe cool down transient. That was the reason for my concern.

MR. DEAVER: Yes. I know we analyzed for that full condition. Obviously, in a real event where we need to cool down, there we account for all four, but we only need three to have some redundancy.

MEMBER CORRADINI: But this, just so we're clear, this will be covered in Chapter 15, and we can return back to this.

MR. DEAVER: Yes.

MEMBER CORRADINI: And I'm sure the members won't forget.

MR. DEAVER: Right. As far as Chapter 5, we started to discuss the materials aspect of the reactor coolant pressure boundary. Probably the area of most interest is on avoiding stress corrosion of stainless steel materials. And we do basically three approaches to controlling material in stainless steel. Basically, the initial materials, we minimize the potential for sensitization by minimizing the carbon content, and preventing heating above 800 degrees Fahrenheit. And we control the heat input controls and fill metals to minimize the potential for stress corrosion cracking.

The second aspect is to control in-process minerals and water quality in order to minimize the contact with contaminate kind of materials, such as fluorides and chlorides, and so forth. Then the last aspect of it is the avoidance of cold work, either in the forming of components, or in the in-process issues associated with the actual fabrication.

We had some discussion last meeting on grinding, and our attempt is to minimize grinding, not to dress up welds after they've been made. If we have to resort to grinding, we will use control processes that will be demonstrated by each of the suppliers ahead of time, such that we make any cold layers that are present.

CHAIRMAN SHACK: Yes. Now just looking at the document, that's really discussed under cold work lost and stainless steels. It's not under the control of welding, which I think was the concern that was raised in the Subcommittee.

MEMBER CORRADINI: I remember Sam saying was to minimize this.

MEMBER ARMIJO: Right. It's the tolerance of an antiquated fabrication technique that is so detrimental to the performance of materials in the BWR, in particular. Let me ask a couple of questions. GE, have you reviewed the incidents of IGSEC on 316 nuclear grade shrouds in Japan? Are you familiar with that event?

MR. FREW: Yes.

MEMBER ARMIJO: Okay. As you recall, that thing was the best material we know how to make for the BWR, and it was badly cracked as a result of heavy post-weld grinding. It also had the issue, it was in the core and all of that, there's some IASEC hardening, but the cold work, the residual tensile stresses on the surface of the material creates a very high susceptibility to crack nucleation. And so the Japanese got in a lot of trouble by that cracking. Cold working, my view is you should be much less tolerant of this, instead of using controls which are very difficult to quantitatively measure. Surface cold work, you're not going to measure that with a hardness test, so I think with the few -- you're going to have fewer welds in the ESBWR. You don't have a recirc system. A lot of these components are being made in shops, weld fabricators, so machine welding should be widely used. So I don't see any reason why you would tolerate processes that are known to nucleate stress corrosion cracks, even on the best materials you have. This is primarily an economic issue, but it eventually can lead to safety issues. And, again, I'd advise you to rethink that, and not be timid about it. Just prohibit it. Just say we won't tolerate it. If you guys -- fabricators can't make good welds without grinding to cover up sloppy work, then get new fabricators, or make them qualify a process that will work. So I just think this is -- you have an opportunity. You're building a new reactor, and there's no reason in the world that you should repeat the same mistakes in the past by being tolerant of antiquated fabrication techniques.

The other question I had was on water quality. I guess I'll broaden it to, hydrogen water chemistry is a reference water chemistry for this system, the ESBWR?

MEMBER SIEBER: Noble metal.

MR. DEAVER: We're designing for it, but it's a choice of each of our customers to decide whether they want to initially operate with hydrogen water chemistry or not. All of our customers, COL applicants at this point are indicating that they're going to use hydrogen at this point.

MEMBER ARMIJO: So GE is designing in hydrogen water chemistry as a reference design?

MR. DEAVER: I would say that we're -- go ahead, Amy.

MS. CUBBAGE: I was going to say, it's the COL applicant action item to decide in the DCD, but the capability has been provided in the design should they choose to use it. But it is not part of the standard certified design.

MR. BEARD: Well, if I can try and clarify.

MEMBER ARMIJO: I don't see why you guys don't insist -

MR. BEARD: All the shielding is done assuming that we're going to have hydrogen water chemistry.

MEMBER ARMIJO: Okay.

MR. BEARD: There are taps and a room

designated for the hydrogen water chemistry skid. The skid is not provided as part of the certified design.

MEMBER ARMIJO: But it's easy for them to use it.

MR. BEARD: Yes.

MEMBER ARMIJO: But you're not providing -- for example, are you providing instrumentation, like ECP instrumentation as part of the design?

MR. DEAVER: We have the capability in the LPRM strings to have that capability, yes.

MEMBER ARMIJO: But it's not part of the reference design. It seems strange to me that all the lessons learned on IGSEC and IASCC in the BWR aren't hardwired into the ESBWR design. And, again, it's for the protection of the materials.

MR. BEARD: I guess we view with our materials controls and process controls that the occurrence is less likely. It's going to be the abnormal kind of cases where, an off-chemistry kind of thing that comes up, that will be the thing that initiates potential cracking in that, so -

MEMBER ARMIJO: Jerry, I've got to disagree with you. IASCC doesn't -- will not be -- your materials won't survive against IASCC without hydrogen, so I don't understand where you are. I just recommend that GEH take a -- do another review of the materials control and the water chemistry to avoid future stress corrosion cracking problems in this plant. And I think the staff should put some pressure on you to do that. With that said, I'll be quiet.

MEMBER CORRADINI: The staff will be up shortly, so I'm sure --

MEMBER ARMIJO: I'm not going to repeat myself since you heard it one time.

(Off the record comments.)

MEMBER ARMIJO: I've spent too many years of my life finding and fixing stress corrosion cracks in the BWRs. And when I see this type of stuff, I can see it coming down -- I see a train wreck coming on a new plant where it's totally unnecessary if you apply the lessons learned from BWR operation.

MEMBER SIEBER: Well, it affects a lot of other things, too. The shielding is different.

CHAIRMAN SHACK: He's built that in.

MEMBER ARMIJO: He's built that in. It just seems like it's the modern way to run a BWR, and why isn't that built into the ESBWR?

CHAIRMAN SHACK: Because the Japanese don't like it.

MEMBER ARMIJO: What's that did you say?

CHAIRMAN SHACK: Japanese don't like it. It's a potential customer, you're not going to -

MEMBER ARMIJO: They can take it out. This is a U.S. -

(Off the record comments.)

MEMBER CORRADINI: We'll move on.

MR. DEAVER: Okay. Next slide. I think that's it.

MEMBER CORRADINI: That might be the last one.

MR. DEAVER: That is the last one. Okay. That's it for Chapter 5, if you have any questions.

MEMBER CORRADINI: Any questions from the members on any of the chapters? If not, thank you very much, and we'll turn it over to the staff. And GEH will be close by to have discussions, as appropriate.

MS. CUBBAGE: I'm going to be making the presentation for the staff today, Amy Cubbage, Lead Project Manager for ESBWR Design Certification. However, I have brought with me a team of the lead reviewers. Sitting with me here is Brad Harvey. He's one of our leads in the Chapter 2 review; Jean-Claude Dehmel, our lead reviewer on Chapter 11, he also has a role on Chapter 12; and Bob Davis, one of our lead reviewers on Chapter 5. Also have many members of the technical staff in the audience. We'd be happy to answer any questions you may have.

So we're going to briefly discuss our safety evaluation report conclusions on Chapters 2, 5, 8, 11, 12, and 17. I'm going to be providing a brief overview of the open items that were contained in the staff's SERs, for some of the chapters that had only a few open items, we'll discuss those. For others that had more open items, I'm going to be presenting a sample of those open items, focusing on the more significant issues.

As you know, we briefed these to the Subcommittee last month. Our safety evaluations are based on GE's Design Control Document Revision 3, and any RAI responses that we had received to-date. DCD Revision 4 was submitted by GE-Hitachi on September 28th, so our SERs do not address that Design Control Document revision. Some additional open items may result in the staff's review of DCD Rev. 4, and we will address those with the Committee when we come back for the final SER.

DCD Revision 5 is expected in March 2008 to address the remaining open issues. We're also looking at the March 2007 SRP which was issued after about a year and a half into this review, for identifying any significant areas where the guidance in the SRP would be reflective of a change that occurred in the regulations, and the impacts resulting from that SRP review will be addressed in the final SER.

MEMBER MAYNARD: I'm sorry. Go back over that again as to what are the -- the Standard Review Plan changed in the middle of this process.

MS. CUBBAGE: That's right.

MEMBER MAYNARD: Well, which one, basically, is the ruling one for the -

MS. CUBBAGE: Well, GE was required to address the SRP in effect six months prior to their application, so that's what the application is based on. And so we're going through the March '07 SRP to look for areas where the acceptance criteria changed to determine if there's an issue that needs to be addressed by GE. GE is required to address all of the regulations in effect at the time of certification, so they don't get finality with the previous version of the SRP. So if there are any areas where we feel that the DCD does not address the current acceptance criteria, we'll ask GE in RAI.

MEMBER MAYNARD: Okay.

MS. CUBBAGE: Okay. For Chapter 2, we have four open items at this time. The first open item is related to the weight of the 48-hour probable maximum winter precipitation. We were concerned that some of that precipitation may fall as frozen precipitation. We also had some questions about the exclusion area boundary chi over Q that was used for some of the event analyses, which is different from the exclusion area boundary chi over Q that was selected as a site parameter. We had some issues with the control room filtered air intake and unfiltered leakage locations, and we had some issues about the assumptions used in driving long-term chi over Q and D over Q site parameters. And Brad would be happy to answer any questions you may have on those.

MEMBER STETKAR: I have one. It's not on those, but does the staff have any problems -- I notice in the DCD that GEH basically screens out any external hazards that are estimated to have a frequency lower than 10 to the minus 7 per year. They're just excluded, and yet, the results from their PRA supposedly accounting for all hazards, during all modes of operation show a total core damage frequency of substantially less than 10 to the minus 7, so that there's a potential that they could be just throwing away things that are greater than the total of everything else that they quantified.

MS. CUBBAGE: I can't address that question.

MEMBER STETKAR: Okay. Thank you.

MS. CUBBAGE: I don't know if GE would like to.

MEMBER CORRADINI: There is somebody there that could address it.

MR. WACKOWIAK: Rick Wackowiak from GE-Hitachi. I think this question came up during the Subcommittee meeting, and -

MEMBER CORRADINI: Yes. I think in Chapter 17 we may have rolled this one over a bit.

MR. WACKOWIAK: It was 17, and 2, possibly.

MEMBER CORRADINI: Yes.

MR. WACKOWIAK: The particular events -- in Chapter 2, what we're looking at is what are the site characteristics, and specifying some things as to what the site should be designed for. The PRA itself, though, does go in and look at other types of events that may fall into this category.

In Section 11 of the PRA, there's a short discussion on the remaining mitigating capability for things like aircraft impact, nearby facility accident, some of these things that we're talking about here. And what we show is that yes, they are small initiating events, but the plant remains with mitigating capabilities that would also have to fail to get to core damage given this event. And our conclusion was that these were not drivers of risk in ESBWR.

MEMBER CORRADINI: Can I ask John's question a bit differently, just because I might have this wrong, and the staff can correct me; which is that, so take one, for example, if there's an external hazard that's outside the five miles by what we had heard on Wednesday, one of these days, was Vogtle, actually, outside the five miles, within the design base for the site characteristic, you don't consider it, but in the PRA you would consider it. Am I getting this correct? That is, if there is some sort of hazard that you would consider for the site characteristic that's not required for site characterization for like an Early Site Permit, or you might have it not in the DCD, but in the PRA it would be considered, and there's no distance aspect to any sort of external hazard that you might want to consider. Am I –

MEMBER STETKAR: That's a little different.

MEMBER CORRADINI: That's a little different? I'm not -

MEMBER STETKAR: Talking about the frequency. A good example is things like meteorites. I don't want to quibble about frequencies, but those events that are judged in one side of the ledger as very, very rare events, and do not merit consideration because of some nominal estimate of some frequency of less than 10 to the minus 7 per year. And, yet, on the other side of the ledger, in the PRA space we're saying that the total core damage frequency is substantially lower than the frequency of those potential hazards.

MEMBER BLEY: There's a chapter, like 18 is on the PRA.

MEMBER CORRADINI: Nineteen.

MEMBER BLEY: Nineteen, and that's -- we haven't done that yet. Right?

MEMBER CORRADINI: No, that's correct.

MEMBER BLEY: That hasn't come up yet, so we can get into those in detail.

MEMBER STETKAR: That's in the PRA space. I'm just asking the staff -

MEMBER BLEY: I understand.

MEMBER STETKAR: -- whether they feel comfortable with that apparent discrepancy.

MS. CUBBAGE: In light of the low CDF for this plant.

MEMBER STETKAR: Right. If the core damage frequency for this plant were 10 to the minus 2, I would never open my mouth. Of course, we wouldn't be sitting here, but that's a different -

MS. CUBBAGE: We are doing our Chapter 2 review consistent with our Standard Review Plan and guidance, and I think that would be a good topic to revisit in the PRA discussion, as to whether there are some external events that could be of a higher frequency than the current CDF.

MR. WACKOWIAK: Could I ask a clarifying question for what it is you're actually asking? Because I think this latest part of the discussion -- I mean, try to see if what I just heard is what you're getting at.

If in the process of designing this plant we eliminated the susceptibility of all previously regulated events, are you suggesting that the staff now create a new set of regulated events that had previously been ignored, but now would be the dominant ones in ESBWR?

MEMBER STETKAR: That's an excellent question, and I'm not sure that I want to comment on that right now.

MR. WACKOWIAK: Okay. Because that last part of your statement is kind of what I heard your question going to, should we address more things in the site characteristics than in other plants, because this plant has a low core damage frequency.

MEMBER STETKAR: I think my question was more to the staff in terms of their sensitivity to an absolute numerical screening criterion or cutoff for the things that they may consider, recognizing that there are certainly events that are beyond what you would normally design a nuclear power plant to withstand. But is the staff sensitive to that absolute numerical cutoff in screening, in the fact that, effectively, the design does not need to consider events with a frequency lower than a specific numerical cutoff, when, indeed, the estimated risk -- I mean, those events may be the lower bound for the total risk, the total core damage frequency, total release frequency in some sense from the plant.

MR. DUBE: Don Dube, Senior Level Advisor, Division of Safety Systems and Risk Assessment in

Office of New Reactors. We had a question along those lines from the AP-1000 Design-Centered Working Group, whereby they are using an approach to address external events on a site-by-site basis. And our response to one of the questions was, well, originally they were proposing to use the IPEEE Guidance, I think it's NUREG-1407, but I could be wrong, regarding screening criteria for external events, whether one would consider those or not. And our response to it was that the screening criteria for external events were based on the current generation of plants which had core damage frequencies of 10 to the minus 5 or so, and that that probably would not be pertinent to the new reactors, which have core damage frequencies of 10 to the minus 8, 10 to the minus 7, and so our response was that one should not apply those screening criteria from the IPEEE Guidance to new reactors, that one should look at the relative risk, the relative core damage frequency for the new reactors, and lower those screening criteria appropriately.

MEMBER STETKAR: Don, is that screening criteria in what I'll call PRA space?

MR. DUBE: In PRA.

MEMBER STETKAR: Okay. Because there is that difference. Rick, I recognize in the PRA you probably expanded that screening appropriately.

MR. WACKOWIAK: Right.

MEMBER STETKAR: Again, I come back to the staff when they're looking at a more deterministic evaluation of hazards at the site.

MS. CUBBAGE: I'm afraid I'm not going to be able to provide any more on that, unless Raul, do you have anything to add?

MEMBER MAYNARD: Well, I don't think we should be asking them to change their screening criteria just because a design that has a much lower core damage frequency. They've got an acceptance criteria, and I think that we may want to talk philosophically about what should the NRC overall do, but for design certification that we're reviewing on this specific case, there's guidance out there, and the staff is bound to abide by that guidance, as well as the applicant, and stuff.

MEMBER CORRADINI: But what I hear, though, is we can return to this when we get to Chapter 19, and we still also have -- we're going to have a Subcommittee meeting about the PRA, the newest version of it, eventually, so we'll pick it up then.

MS. CUBBAGE: In the Chapter 19 context, yes.

MEMBER STETKAR: Right. Chapter 19 space. Okay.

MS. CUBBAGE: So for Chapter 5, we've issued a total of 138 RAIs, and now 118 have been resolved. These are some of the key open items we have remaining. The first one, ASME code case, the use of ASTM A709 HPS 70W materials. Bob is here to help me out if you have any questions on that. The use of ASTM A800 versus Hulls Equivalent Factors delta ferrite content and cast austenitic stainless steels. On that issue, I understand -

CHAIRMAN SHACK: Just on that first one, the real issue as I took away from the Subcommittee meeting, was that the code case wasn't done yet. Nobody has a real disagreement that this isn't a reasonable material to use.

MR. DAVIS: I think the issue is, is that it's quenched and tempered, so it can't be post-weld heat treated. Division 2 requires when you attach is to the liner, you have to post-weld heat treat it, so I think the material itself used for internal structures is allowed by N690 ANC ISC. It's fine to use. The issue is, is that if they do not post-weld heat treat the liner, what is the affect on that material, not the 709 that they're going to connect to it. And I think the code in GE has quite a bit of work to do to provide us with -- we've just realized that they were going to -- it kind of just came out last month.

CHAIRMAN SHACK: I see that they were actually going to weld it to the shell. Is that -

MR. DAVIS: Well, we don't really even have all the configurations and joint designs that they would use, but what I'm imagining is that, obviously, if you have an in-bed plate, and the liner welds onto that in-bed plate, the in-bed plate becomes the liner, so you have to weld to the in-bed plate with your internal structures. And how thick those are, and what configurations are, what joint designs there, I have no idea. But we do have an RAI question asking that, so we'll be looking at that; plus, they will also be required to explain why it's okay to not post-weld heat treat heavy sections of the liner.

MS. CUBBAGE: The second item on Hulls Equivalent Factors, we have tentative agreement at this time that GE is going to use the Hulls Equivalent Factors, is my understanding. Bob, would you like to elaborate on that one at all?

MR. DAVIS: I mean, other than they've agreed to use that, and that is consistent with the staff's guidance.

MS. CUBBAGE: Okay. The third item, component accessibility for inspections per the ASME Code Section 11, and 10 CFR 50.55(a). The concern there was that we want to make sure that all of the welds are designed for inspectability through either ultrasonic examination or radiography, with the expectation that we do not receive relief requests in the future due to the design of these components.

We asked a number of RAIs related to materials selection, and the integrity of the isolation condenser and PCCS tubes. We're waiting for GE's response on those. We have an open issue regarding the sensitivity and alarm limit for unidentified reactor coolant pressure boundary leakage. The staff was concerned that the 5 gpm tech spec limit was too high, such that we needed to have some assurance that there would be operational controls and procedures to take corrective actions on low levels of unidentified leakage. And, lastly, we had some issues regarding the capability of the reactor water cleanup shutdown cooling system for decay heat removal. These issues related to the reactor pressure vessel level required. The inlet and outlet could create a condition where we're not getting adequate flow through the core, so we're waiting for a response from GE on that question, as well.

CHAIRMAN SHACK: On this component accessibility, I mean, you're essentially asking for all the welds to be inspectible.

MR. DAVIS: Well, what our concern is, is that not all of these components, in particular, we're concerned about the austenitic welds, which would be austenitic to austenitic and dissimilar metal welds, that if you cannot access those from both sides, you can't do -- you can only do an ultrasonic examination from the side that can be examined. Therefore, the code does allow radiography to be used, and our concern is a lot of times licensees come in with relief requests to not perform radiography due to ALARA issues. Obviously, you have to drain everything. You can't have water in the pipe if you're going to perform radiography, so our position is, is that if you -- you have to make these welds so that they're practical to be examined. So if you say that you're going to RT it, you're going to RT it when the time comes, and that there won't be any relief request associated with impracticality.

MEMBER ARMIJO: Let me ask you a question on this new component, this chimney, big component with lots of things. Are they -- I don't know if enough detail has already been done on it, but are those welded together full length along every one of those channels, and how do you inspect those?

MR. DAVIS: I have not -- I am not involved with the reactor vessel internals in any way, shape, or form, so I -

MEMBER ARMIJO: But the issue is -

MR. DAVIS: I'll have to divert that question to someone who's familiar with reactor vessel internal inspections. And that's -- I believe reactor vessel internals is not part of Section 11, which is what this section covers.

MS. GRUSS: This is Kim Gruss, Chief of Component Integrity Performance and Testing. The issue what you're describing or discussing is part of the Chapter 4.5.1 Section, so we will have an opportunity to discuss it.

MS. CUBBAGE: I don't know whether GE would like to take an opportunity to explain the fabrication.

MR. DEAVER: Yes. Our intent is to -- this is Jerry Deaver. Our intent is to weld full length the partitions, and we have a cruciform design to help facilitate making those welds at the junctions, so we'll provide you a lot of detail when we get to that part of it.

MEMBER ARMIJO: But it would have to meet the inspectability criteria of -

MR. DEAVER: Yes.

MR. DAVIS: This is Section 11.

MEMBER ARMIJO: Section 11.

MR. DAVIS: Reactor vessel internals would be covered under something else, other than Section 11.

MEMBER ARMIJO: Okay.

MEMBER CORRADINI: I think we should wait on hearing about this.

MEMBER STETKAR: I have a question on Chapter 5. Where in the SER have you examined or commented on the main steam isolation valves?

MS. CUBBAGE: The valves? They're not part of this chapter. I believe they're part of Chapter 3.

(Off the record comments.)

MEMBER CORRADINI: What is the name of the -- are the main steam isolation -- since they're included in the DCD Chapter 5.

MS. CUBBAGE: Okay.

MEMBER CORRADINI: They're not in your Chapter 10, because your Chapter 10 refers to Chapter 5 of the SER. Your SER Chapter 10 refers to Chapter 5 of the SER for the main steam and main feedwater isolation. But I couldn't find it in Chapter 5 of the SER, so I'm curious where they are.

MS. GRUSS: This is Kim Gruss, again. The information on that is in Chapter 3, so the pumps and valves design and inspection and testing are in Chapter 3.

MEMBER CORRADINI: Okay.

MS. CUBBAGE: All set on Chapter 5? Okay. Chapter 8, we had 116 RAIs, and all but one at this point are resolved. This remaining open item has to do with sizing of the batteries. And, actually, at this time we do have a path forward on this. GE has verbally committed to provide the sizing, and the load for the batteries in DCD Tier 2, so we're waiting for that response to be submitted.

Chapter 11, out of the original 88 RAIs, we have three remaining open items. They all have a very similar thread with the mobile systems, as they were referred to in DCD Revs 3 and 4. GE intends to rename those in the future DCD revision to be process sub-systems, getting away from that term "mobile" because it wasn't really reflective of the nature of these systems that are not going to be brought in and out on a daily basis. And, also, that GE plans to identify these sub-systems as certified design material in the next DCD revision, so they will not be conceptual. Any questions on Chapter 11?

Okay. Chapter 12. At this time there are 24 remaining open items. I'm just going to summarize a few of them. We had some questions about the source term, how it was estimated for the airborne and liquid effluent releases. An open question about the adequacy of the shielding for the inclined fuel transfer tube area to insure that while fuel is being transported, that there's no possibility that someone could be in that vicinity, and that there's adequate shielding. We had open RAI on the post accident dose rates near the HVAC filters. A number of open RAIs related to the location of vital areas on post accident radiation zone drawings, and associated post accident mission dose for these areas. RAIs related to the dose assessment for operational exposure. This relates to the estimates for certain maintenance activities. And, lastly, RAIs related to conformance with 10 CFR 20.1406 related to the minimization of contamination.

Okay. On Chapter 17, we have one remaining open item. It's been discussed here earlier this morning, and we have a path forward on that item. We're just waiting for GE to submit the results of their expert panel, identifying the risk-significant SSCs within the scope of the D-RAP program.

So, in conclusion, we are requesting feedback from the ACRS on all of these chapters we've presented today based on the SER with open items and DCD Revision 3. We will be briefing the Committee on the final SER in late 2008, and early 2009. At that time, we'll address the closure of the open items, and address any changes resulting from DCD revisions.

In the near term, we have a meeting planned on November 15th to discuss Chapters 9, 10, 13, and 16. Those SERs have been provided to the Committee. Chapter 10 was provided a few weeks back, and Gary now has Chapters 9, 13, and 16, and he'll be distributing those today, I believe. And the next interaction we're planning for in January would be to address Chapters 4, 6, 15, and 21.

MEMBER CORRADINI: If I could just take a minute. These are what one might call beefy chapters, and so it's the intent in speaking with -- Gary and I, talking with Amy, that we would probably want to dedicate two full days of Subcommittee to discuss these, because, if I remember correctly, four is auxiliary systems, six is engineering -

MS. CUBBAGE: Four is Reactor Internals -

MEMBER CORRADINI: Reactor Internals, excuse me. And six is Engineering Safety Features, 15 is the Accident Analysis, and I don't remember 21.

MS. CUBBAGE: Twenty-one is where we address the methods for the accident analysis, or the TRACG review. The qualification testing supporting those, the -

MEMBER CORRADINI: Sanjoy is getting excited. But in any case, I think this is an important grouping that we want to do together, because a lot of these things are inter-dependent.

MS. CUBBAGE: So that concludes my presentation. If there are any additional questions, we'd be happy to -

MEMBER ABDEL-KHALIK: Now this follow-up review of the final SER in late '08 or early '09, is that a time where the entire SER will be reviewed in toto, so that --

MS. CUBBAGE: The entire --

MEMBER ABDEL-KHALIK: You know, my concern really with the process of reviewing things on a chapter-by-chapter basis is that things do fall between the cracks. And many of the questions that come up really come up at the interface between different chapters.

MS. CUBBAGE: Right. At the final SER stage, we'll be presenting you with an entire SER at one time. We're allowing in our schedule for multiple interactions, because we recognize it's going to be a very large document, so we're allowing for two Subcommittee meetings, and two Full Committee meetings in that time frame.

MEMBER ABDEL-KHALIK: So what would a letter commenting on these six chapters mean?

MS. CUBBAGE: Right. What we're looking for is, if you have any comments that we need to factor into the review, we're looking for that. We're also looking, if you don't have any comments on the degree of finality, that you approve of what the staff has done to-date, if any additional questions come up later, you're welcome to send those to the staff at future meetings.

MEMBER CORRADINI: I think the impression that I got in speaking with the staff is that we are not precluded to going back if we find a system interaction issue that brings us back. For example, if there's something in, I'll pick electrical power, that somehow is involved with some other thing, you're going to have -- if you have to go back and discuss it, we have to go back and discuss it.

MEMBER ABDEL-KHALIK: Okay. Thank you.

MEMBER CORRADINI: Any other questions by the members? Mr. Chairman, we could discuss our feelings about the chapters now. We could turn it back to you and discuss it later after break. I leave it in your capable hands.

CHAIRMAN SHACK: Let's discuss it in the context of the letter, I think.

MEMBER CORRADINI: Okay. Now?

CHAIRMAN SHACK: No, let's put that off until later.

MEMBER CORRADINI: Okay. That's what I wanted to know, when you wanted to do it, or how you want to do it. That's fine.

CHAIRMAN SHACK: At the moment, we're ahead of schedule.

MEMBER CORRADINI: But before we jump to potentially a break, I wanted to thank GEH and the folks that came back today to help answer the questions, thank the staff, and Amy, in particular, because staff has been very good at getting us things enough in advance that, at least it's in our camp to read it and appropriately digest it, so thanks to both. I really appreciate their time and efforts.

CHAIRMAN SHACK: Okay. I'm going to declare a break until 10:15. We were ahead of schedule, but the upcoming things are basically internal things, so we can, I think, go ahead and start working on those.

(Whereupon, the proceedings went off the record at 9:53 a.m.)

CERTIFICATE

This is to certify that the attached proceedings before the United States Nuclear Regulatory Commission in the matter of:

Name of Proceeding: Advisory Committee on Reactor Safeguards

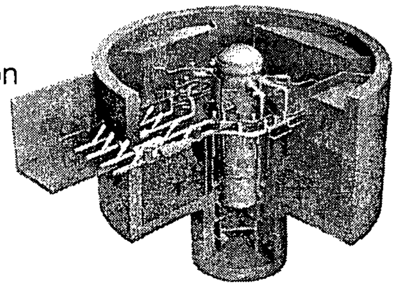
Docket Number: n/a

Location: Rockville, MD

were held as herein appears, and that this is the original transcript thereof for the file of the United States Nuclear Regulatory Commission taken by me and, thereafter reduced to typewriting by me or under the direction of the court reporting company, and that the transcript is a true and accurate record of the foregoing proceedings.

Charles Morrison
Official Reporter
Neal R. Gross & Co., Inc.

ESBWR - Overview
DCD Chapters 2, 5, 8, 11, 12, and 17



Advisory Committee on Reactor Safeguards
November 2, 2007
Jim Kinsey
Alan Beard
Rick Wachowiak
Jerry Deaver
Brian Frew

GE Hitachi Nuclear Energy

Presentation Content

- Introduction
- Overview of SER Chapters 2, 8, 11, 12, 17 and 5

Introduction

- Select DCD Chapters and associated draft NRC Safety Evaluations presented to ACRS Subcommittee.
  - > Chapters 2, 8 and 17 on October 3, 2007
  - > Chapters 5, 11 and 12 on October 25, 2007
- Brief overview of the above DCD Chapters will be presented.

Overview of Chapter 2 – Site Characteristics

- Chapter 2 provides description of:
  - > Meteorology
  - > Hydrology
  - > Geology
  - > Seismology
  - > Geotechnical Parameters
  - > Potential Nearby Hazards
- The applicant referencing the ESBWR DCD will:
  - > Establish the actual site characteristics
  - > Demonstrate site parameters

Overview of Chapter 8 – Electrical Power

- Chapter 8 provides description of:
  - > Offsite Power Distribution
  - > Onsite Power Distribution Including
    - – Plant Investment Protection Buses
    - – Safety-Related AC / DC Power Supplies
  - > Station Blackout (SBO) Analysis Provided in Chapter 15

Overview of Chapter 11 – Radioactive Waste Management

- Chapter 11 describes the radioactive waste streams; how they are processed, monitored and sampled; and radiation monitors that initiate safety-related functions.
  - > Source Terms – Fission / Activation Products
  - > Liquid Waste Management
  - > Gaseous Waste Management
  - > Process and Effluent Monitoring and Sampling

Overview of Chapter 12 - Radiation Protection

- Chapter 12 describes administrative programs and procedures, in conjunction with facility design, to ensure that the occupational radiation exposure to personnel will be kept ALARA.
  - > Radiation Sources
  - > Radiation Protection
  - > Dose Assessment
  - > Health Physics
  - > Minimization of Contamination and Waste Generation

Overview of Chapter 17

- Chapter 17 provides description of:
  - > GEH QA Program Description that establishes the Quality Assurance requirements implemented during ESBWR design.
  - > GEH ESBWR work control process defines the supplier and sub-tier supplier quality program requirements.
  - > GEH QA responsibilities.
  - > ESBWR Design Reliability Assurance Program (D-RAP)
    - Assures important ESBWR reliability PRA assumptions are considered throughout plant life
    - Includes risk-significant SSCs that provide defense-in-depth or result in significant improvement in the PRA
    - The site specific D-RAP will be confirmed following construction and will be verified using ITAAC

Overview of Chapter 5 - RCS and Connected Systems

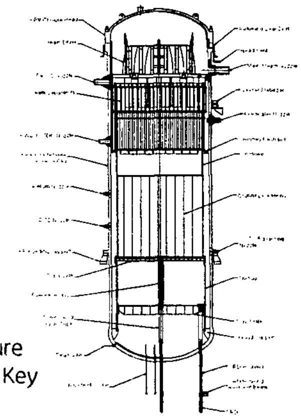
- Chapter 5 provides description of:
  - > Reactor Coolant System (RCS) including those systems and components that contain or transport fluids coming from or going to the reactor core
    - These form the major portion of reactor coolant pressure boundary (RCPB)
  - > RCPB components up to and including:
    - Outermost containment isolation valve in piping that penetrates containment
    - Second of two valves normally closed during normal operation
    - RCS safety/relief valve (SRV) piping and depressurization valve (DPV) piping

Overview of Chapter 5 - Reactor Vessel

ESBWR RPV Changes from Prior BWRs

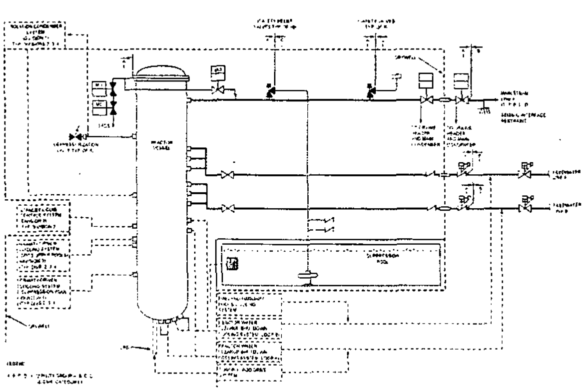
- > Major Reactor Vessel Penetrations Are Above Top of Active Fuel
- > 6 Large Ring Forgings Are Used; Closure Flanges and 4 Lower Vessel Forgings Including Core Beltline Region
- > Core Region Is ~2 Ft Shorter Due to Shorter Fuel
- > Chimney Is a New Component to Facilitate Natural Circulation
- > RPV Height Is ~ 6.5 M Higher Than ABWR
- > RPV Diameter Is 7.1 M - Same As ABWR
- > RPV Head Vent Exits from Main RPV Body
- > Main Steam Flow Restrictor Is Integral With RPV Nozzle
- > RPV Supported by 8 Sliding Supports

Overview of Chapter 5 - Reactor Vessel

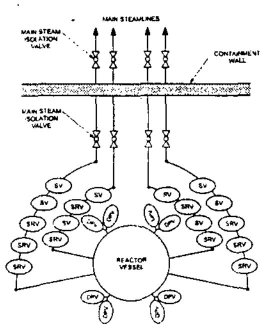


Reactor Pressure Vessel System Key Features

Overview of Chapter 5 - Nuclear Boiler System

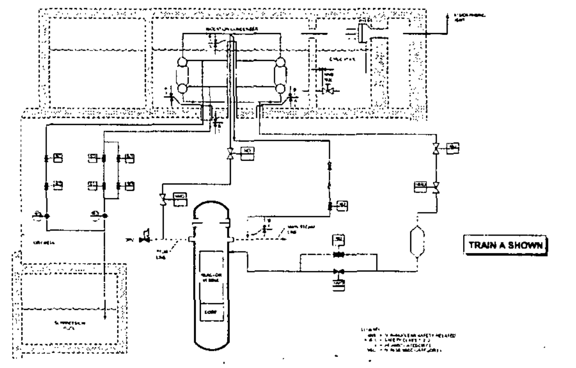


Overview of Chapter 5 – Integrity of RCPB



Safety-Relief Valves, Safety Valves, and Depressurization Valves on Steamlines

Overview of Chapter 5 – Isolation Condenser System



Overview of Chapter 5 – Integrity of RCPB

Measures to Avoid Stress Corrosion Cracking in SST

- Avoidance of sensitization
  - Materials supplied in solution heat treated (SHT) condition
  - Carbon Content < 0.02%
  - No heating above 800°F during fab unless SHT applied
  - Welding heat input controls and filler metal ferrite control
- Process controls to minimize contaminants during fabrication
  - Control of process materials and water quality
  - Cleanliness prior to elevated temperature treatment
- Avoidance of Cold Work
  - Controlled by applying hardness limits
  - Surface finish and process control on ground surfaces

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Protecting People and the Environment

Presentation to the ACRS
ESBWR Design Certification Review
Chapters 2, 5, 8, 11, 12 and 17

Amy Cubbage, Senior Project Manager, NRO
November 2, 2007

ACRS Presentation
ESBWR Design Certification Review

Purpose:

- Brief the ACRS on the ESBWR Design Certification review
- Summarize the open items contained in the Staff's SER with Open Items for Chapters 2, 5, 8, 11, 12 and 17
- Address the Committee's questions on these Chapters

ACRS Presentation
ESBWR Design Certification Review

Subcommittee Briefings:

- October 2 and 3, 2007
  - Design overview and project status
  - SER with open items for Chapters 2, 8 and 17
- October 25, 2007
  - SER with open items for Chapters 5, 11 and 12

ACRS Presentation
ESBWR Design Certification Review

Background:

- SERs based on DCD Rev. 3 and RAI responses
- DCD Rev. 4 submitted by GEH on September 28, 2007
  - Some open items may be resolved by DCD Rev. 4
  - Additional RAIs expected based on the changes in DCD Rev. 4
- DCD Rev. 5 expected in March 2008 – to address remaining open items
- Staff identifying any changes in the March 2007 SRP that would impact the staff's conclusions
  - Any impacts will be addressed in the final SER

ACRS Presentation
ESBWR Design Certification Review

Chapter 2 RAI Status Summary

- Original RAIs: **54**
- RAIs resolved: **50**
- Open Items: **4**

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ACRS Presentation
ESBWR Design Certification Review

Chapter 2 Open Items:

- Provide an additional roof design site parameter to account for additional weight if at least part of the 48-hr probable maximum winter precipitation falls as frozen precipitation. (Open Item 2.3-4)
- Discuss why a EAB X/Q value of $1x10^{-3}$ used in the feedwater line break and RWCU/SDC line break accidents differs from the EAB X/Q site parameter of $2x10^{-3}$. (Open Item 2.3-8)
- Identify the control room filtered air intake and unfiltered inleakage locations and potential release pathways to the environment for each accident. (Open Item 2.3-9)
- Discuss the assumptions used in deriving the long-term average X/Q and D/Q site parameters. (Open Item 2.3-10)

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ACRS Presentation
ESBWR Design Certification Review

Chapter 5 RAI Status Summary

- Original RAIs: 138
- RAIs resolved: 118
- Open Items: 20

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ACRS Presentation
ESBWR Design Certification Review

Chapter 5 Open Items:

- ASME Code Case – use of ASTM A709 HPS 70W materials (Open item 5.2-50)
- Use of ASTM A800 vs. Hulls Equivalent Factors for delta ferrite content in cast austenitic stainless steels (Open item 5.2-38)
- Component accessibility for inspections per ASME Code, Section XI and 10 CFR 50.55a (Open item 5.2-62)
- Issues related to materials selection and integrity of ICS and PCCS tubes (Open items 5.4-20, 5.4-53, 5.4-55 through 58)
- Instrument sensitivity and alarm limit for unidentified RCPB leakage (Open items 5.2-2 and 16.2-4)
- Issues related to RWCU/SDC decay heat removal capability (RPV level required, flow and mixing) (Open item 5.4-59)

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ACRS Presentation
ESBWR Design Certification Review

Chapter 8 RAI Status Summary

- Original RAIs: 116
- RAIs resolved: 115
- Open Items: 1

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ACRS Presentation
ESBWR Design Certification Review

Chapter 8 Open Item:

- Staff requested GEH to provide the loading profile on UPS (i.e., battery sizing) (Open Item 8.3-52)

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ACRS Presentation
ESBWR Design Certification Review

Chapter 11 RAI Status Summary

- Original RAIs: 88
- RAIs resolved: 85
- Open Items: 3

ACRS Presentation
ESBWR Design Certification Review

Chapter 11 Open Items:

- Three RAIs remain open
  - Staff has requested ITAAC be added for the solid waste management system (Open Item 11.4-15)
  - Staff requested clarification of the conceptual nature of the mobile liquid waste management system (LWMS) design in the DCD. (Open Item 11.2-16)
  - Staff requested clarification of the conceptual nature of the mobile solid waste management system (SWMS) design in the DCD. (Open Item 11.4-18)

ACRS Presentation
ESBWR Design Certification Review

Chapter 12 RAI Status Summary

- Original RAIs: 80
- RAIs resolved: 56
- Open Items: 24

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ACRS Presentation
ESBWR Design Certification Review

Chapter 12 Open Items:

- Estimate of the source term for airborne and liquid effluent releases (Open Items 12.2-9 and 12.2-15)
- Adequacy of shielding for inclined fuel transfer tube area (Open Item 12.4-19)
- Description of post-accident dose rates near HVAC filters (Open Item 12.4-23)
- Location of vital areas on post-accident radiation zone drawings and associated post-accident mission doses for these areas (Open Items 12.4-31, 12.4-32, and 12.4-33)
- Issues related to dose assessment for operational exposures (Open Items 12.5-1, 12.5-6, and 12.5-8)
- Issues related to conformance with 10 CFR 20.1406 (minimization of contamination) (Open Items 12.7-1, 12.7-2, and 12.7-3)

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ACRS Presentation
ESBWR Design Certification Review

Chapter 17 RAI Status Summary

- Original RAIs: 19
- RAIs resolved: 18
- Open Items: 1

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ACRS Presentation
ESBWR Design Certification Review

Chapter 17 Open Item:

List of risk significant SSCs within D-RAP (Open Item 17.4-1)

- Staff requested that GEH identify the risk-significant SSCs within the scope of the D-RAP.
- GEH is assembling an expert panel with experts in probabilistic risk assessment, engineering judgment and operating experience to identify the list risk significant SSCs within the scope of D-RAP for the ESBWR design.

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ACRS Presentation
ESBWR Design Certification Review

Conclusions:

- The staff is requesting feedback from the ACRS on Chapters 2, 5, 8, 11, 12, and 17
  - Based on SER with open items and DCD Rev 3
- The staff will brief the ACRS on the final SER
  - Late 2008/Early 2009
  - Address closure of open items
  - Address changes resulting from GEH revisions to DCD
- Additional meetings planned near term
  - SER with open items Chapters 9, 10, 13, and 16
    - November 15 Subcommittee/December 6 or 7 Full Committee
  - SER with open items Chapters 4, 6, 15 and 21
    - Late January Subcommittee/February Full Committee
  - Remaining Chapters Spring 2008

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Backup

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ACRS Presentation
ESBWR Design Certification Review
Site Parameters

- A design certification applicant provides postulated site parameters for the design, and an evaluation of the design in terms of such parameters
- DCD Tier 1 and 2 define the envelope of site-related parameters that the ESBWR Standard Plant is designed to accommodate
- The specified site parameters are the top-level bounding site parameters used to define a suitable site for a facility referencing the certified design
- COL applicants referencing a certified design are required to demonstrate compliance with the site parameters provided in DCD Tier 2, Table 2.0-1