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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

October 25, 2007

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

ESBWR SUBCOMMITTEE

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MEETING

+ + + + +

THURSDAY,

OCTOBER 25, 2007

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The meeting was convened at 8:30 a.m. in room T2B3 at Two White Flint, NRC Headquarters, 11545 Rockville Pike, Rockville, Maryland, Michael Corradini, Subcommittee Chairman, presiding.

<u>MEMBERS PRESENT</u>:

MICHAEL CORRADINI, Chair
WILLIAM SHACK
DANA A. POWERS
J. SAM ARMIJO
SAID ABDEL-KHALIK
OTTO L. MAYNARD

CONSULTANTS TO THE SUBCOMMITTEE PRESENT:

THOMAS S. KRESS

NRC STAFF PRESENT:

AMY CUBBAGE

ANDREA JOHNSON

JEAN-CLAUDE DEHMEL

MOHAMMED SHUAIBI

ILKA BERRIOS

CHARLES HINSON

ERIC OESTERLE

GEORGE THOMAS

ROBERT DAVIS

CHANG LI

NEIL RAY

JAI LEE

ALSO PRESENT:

JIM KINSEY
DALE McCULLOUGH
ERIK KIRSTEIN
FROSTIE WHITE
HUGH UPTON
BRIAN FREW
JOEL MELITO
JERRY DEAVER
JEFF WAAL
LARRY TUCKER

I-N-D-E-X

P R O C E E D I N G S

CHAIR CORRADINI: Let's get started. This is a meeting of the ESBWR Subcommittee. My name is Mike Corradini, chair of the subcommittee. Other ACRS members in attendance are Said Abdel-Khalik, Sam Armijo, Otto Maynard, Dana Powers and Bill Shack. Tom Kress is also attending as a consultant to the subcommittee. Gary Hammer of the ACRS staff is a designated federal official for this meeting.

The purpose of this meeting is to review and discuss the safety evaluation report with open items for several chapters of the ESBWR design cert. We will hear presentations from NRC's Office of New Reactors, GE Hitachi Nuclear Energy Americas LLC.

The subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full committee.

The rules for participation in today's meeting have been announced as part of the notice of this meeting, previously published in the Federal Register.

Portions of this meeting may be closed for the discussion of unclassified safeguards and proprietary information.

We received no written comments or requests for time to make oral statements from members of the general public regarding today's meeting.

A transcript of the meeting is being kept and will be made available as stated in the Federal Register notice.

Therefore, we request that participants in the meeting use the microphones located throughout the meeting room when addressing the subcommittee.

The participants should first identify themselves and speak with sufficient clarity and volume so that they may be readily heard.

We'll now proceed with the meeting and I'll call upon Jim Kinsey of GE-Hitachi Nuclear Energy Americas to begin. Jim.

MR. KINSEY: Thank you. I'm Jim Kinsey. I'm the vice president of ESBWR licensing at GE-Hitachi. I just wanted to take a moment to thank the committee for our first session a couple of weeks ago. We think that this format, covering chapter safety evaluation inputs on a piece-part basis, they have been very efficient and helps us to focus on open issues and close them again most effectively, so we appreciate that process.

Today, we're planning to present, as you know, on the agenda chapters 11 and 12 this morning, and then move on to chapter five as we finish those first two.

We've done a little bit of restructuring since our first session, just to again promote efficiency, and our team will be presenting primarily an overview of key design features or design issues associated with the ESBWR, with a very brief summary at the end NRC or SCR open items, and then we'll turn that over to the NRC staff to go into those issues in more detail.

So Frostie, if you want to introduce the team.

MR. WHITE: Good morning. I'm Frostie White. I'm the lead licensing engineer for both Chapter 11 and Chapter 12 on solid waste process and effluent monitoring and radiation protection.

I'd like to introduce my colleagues. Dale McCullough who's our Chapter 11 solid waste and process and effluent monitoring lead engineer.

And Mr. Kirstein, our Chapter 12 radiation protection engineer.

We're going to begin with Chapter 11 and let you know, both of these individuals will be here for both presentations because both of those chapters intertwined together.

Dale.

MR. McCULLOUGH: Hi. As Frostie said, my name is Dale McCullough. I work for GE-Hitachi. I'm the lead rad waste engineer for Chapter 11.

Our presentation will start with an overview, design parameters, and applicable references and finish up with a summary board which we'll be turning over to the staff.

Chapter 11 describes all the radioactive waste systems in the plant, discusses how waste is processed, the source terms, and the radiation monitors which are used to monitor the process within the plant and the effluents that are released from the plant. 11.1 discusses the source term. 11.2 is liquid waste management. 11.3 is gaseous waste management, formerly off-gas for BWR. 11.4 is solid waste management. 11.5 is process effluent monitoring, sampling, which includes ODCM.

Okay. The first thing I'll go over is what, the col items. So an applicant referencing ESBWR, DCD, will have to ensure that the liquid mobile portable system will comply with Reg Guide 1.143, will identify the interfaces with nonradioactive systems, so that the guidance of 8010 is incorporated.

We'll describe all the procedures and implementation for the mobile portable system, so that we will minimize the waste generation and facilitate, ultimately, decommissioning.

And also have to provide a process control program, which is typical of what's existing in the plants at this time already.

It'll have to provide a plan for temporary storage, if one is to be established, and as part of 11.5, we'll have a lower limit of detection for effluent monitoring systems, develop an off-site dose calculation manual, and develop a--show in the ODCM that the doses for gaseous and liquid effluents will be in accordance with 10CFR 50, Appendix I, and then also provide instrument sensitivities for the instruments that will do this function.

As part of the design, the systems are going to have backup capability, so that you'll be able to perform maintenance and still not limit the processing capabilities. Once again, it will be designed in accordance with Reg Guide 1.143. Factor in operating experience from BWRs right now. That's going to minimize spread of contamination, reduce waste generation and minimize effluent releases. ALARA will be factored into the design.

Some of the design parameters you see on the screen there, the pertinent parts of 10CFR 50--or 10CFR 20, and part 61 for burial, 141.94, environmental radiation doses.

The source term calculations in 11.1, now they use the ANSI standard, 18.1. The design basis, noble gas release as you see on the screen. Design basis iodine, source term, based on iodine 131, we create. And the source terms support the analysis for Chapter 12 and Chapter 15.

Okay. We're doing the liquid rad waste. Liquid rad waste is typical of existing BWRs where we have waste, effluent waste stream segregation, so that the low conductivity waste, high conductivity waste, chemical and turbine wastes are processed in a way that's most efficient.

And the process equipment is similar to existing BWRs. We have filters to remove insoluble, and demineralizers, or reverse osmosis units to remove the insoluble contaminants.

Sample tanks collect the batches of waste. They're sampled and processed. Primarily back to condensate storage or to the environment, provided release limits are met, and the plant inventory demands that.

We're going to use, in the BWRs, skid-mounted equipment, which is lessons learned from the existing plants, to allow us to use the best available processing that's in vogue at the time the plants are being built. And this will reduce generation, waste generation, afford improved maintenance compared to the current designs that are in the plants these days.

Unlike the existing plants, where we had equipment that was, turned out not to be as efficient as we later learned, we're going to have mobile equipment, so that we'll avoid equipment that's going to create a lot of maintenance and high dose.

MR. KRESS: How does the mobile --

MR. McCULLOUGH: What's that?

MR. KRESS: How does the mobile, the company staff compare to fixing --

MR. McCULLOUGH: Oh. The change of equip--the equipment will have a portable shield, removable shielding. It will be designed, right off the bat, to facilitate maintenance and ease of removal, if we have to replace, say, the whole skid and we find out this is not an efficient way to process waste.

MR. KRESS: You can throw the whole thing away.

MR. McCULLOUGH: Yes; that's correct. And we might find--we've found, over the years, better ways to process waste. You know, we get away from things like evaporators, that have been high maintenance, high dose problems.

MR. KRESS: Have you had experience with the mobile units before?

MR. McCULLOUGH: Yes. Prior to joining GE, I was at Exelon, and in my station we didn't because we didn't have the room, but other stations, they used the mobile equipment from different vendors and it's been very successful. They have, you know, reduced--you know, improved water quality, ended up generating less res in the process, same amount of liquid, and has been sort of streamlined as far as--

MR. KRESS: Thank you.

MEMBER MAYNARD: But mobile skids, these are primarily skids that are in locations, that you can remove the whole skid. It's not--

MR. McCULLOUGH: Right.

MEMBER MAYNARD: By mobile, sometimes it sounds like you just move it around and stuff. But it's pretty much in place. But it's easy to remove and replace with another skid, if you needed to do that.

MR. McCULLOUGH: That's correct, sir.

Mobile was the term that was given earlier on, that you see in the EPRI texts, and people think of mobile as something that's on the back of a tractor-trailer, when, in fact, it's a substantial skid with shielding. It's mobile like a condensate pump is mobile. But it's easily removed and set up there to be able to-- you know, with the ability to change.

CHAIR CORRADINI: Further questions?

MR. KRESS: Is that part of the design and control document, or is that left to the COL to decide what they want?

MR. McCULLOUGH: Well, at present, we have it as the--we have it as conceptual information. We're considered, right now, to have that as, we're going to assume, in a next revision to make that the permanent design. But as we talk right now, we're looking at D CD in the current Rev 3, and it's going to, you know, have it--in Rev 4 actually it's shown as conceptual but our plan is to make that--

MEMBER MAYNARD: Is your plan to use existing mobile skids, or is this something that would be designed as part of, unique to this facility?

MR. McCULLOUGH: Well, the idea was that different utilities may prefer one vendor over another, that by having a mobile skid at one vendor with one utility--you're familiar with Energy Solutions. For example, they could use their equipment and out-system against advanced liquid processing system or a Thermex for an RO. Someone else may want a diversified technology. So that way, it would give the utility the flexibility to use the skid, the equipment that they want, the vendor, preferred vendor. That, you know, the requirements for decontamination factors and Reg Guide 1.143 would have to be followed, you know, or specified when they procure that equipment.

It's designed for total recycled liquid radwaste, designed for ALARA to minimize the spread of contamination and facilitate decommissioning, and as I said before, we want to utilize the best processing equipment available and avoid use of equipment which is high maintenance such as evaporators, high maintenance and high dose.

The offgas system is typical of existing BWRs. We have a hydrogen/oxygen recombination moist removal and then hold-up and decay in charcoal base.

MR. KRESS: Are these places that there've been hydrogen explosions, these offgas lines?

MR. McCULLOUGH: The offgas system is designed to--I mean, the explosion, the transient is factored in the design, the calculation that supports the piping would be able to withstand--

MR. KRESS: Piping would be able to withstand it.

MR. McCULLOUGH: Now the offgas system, it's a robust system that's capable of processing three times the source term, mechanically processing three times the source term without affecting delay time of noble gases.

And it's based on a conservative analysis, and just because of the source terms being--that are provided are very conservative.

MR. KRESS: Is this the non-barrier fuel that you're using?

MR. McCULLOUGH: I'd like to defer that to the fuel--

CHAIR CORRADINI: Say it again, Tom. I'm sorry.

MR. KRESS: I was wondering if they were using the non-barrier fuel and whether they had much experience with the leak rates from that. But we can worry about that--

CHAIR CORRADINI: We can defer it.

MR. McCULLOUGH: I can defer that question to--

MEMBER ARMIJO: Would that be in Chapter 5, the fuels?

MR. McCULLOUGH: No.

MS. WHITE: It's going to be in Chapter 4.

MR. McCULLOUGH: Chapter 4.

MR. KRESS: So we're not going to talk about that today.

MEMBER ABDEL-KHALIK: Where do the design basis numbers on slide A come from?

MR. KRESS: That was kind a my question too.

MR. KIRSTEIN: Okay. Those are historic, GE historic design basis for noble gas release rates, the 100,000 microcuries per second. There's a GE document, I believe it's NEDO 10.871, that addresses that.

MR. KRESS: This is based on experience?

MR. KIRSTEIN: Yes.

MEMBER ABDEL-KHALIK: Experience. What kind of experience?

MR. McCULLOUGH: Well, it's actually quite a historic number. I think it's even back from--this document was generated back in the early '70s.

MR. KIRSTEIN: They back-calculate these numbers from the activity they measure in the coolant, in places?

MR. McCULLOUGH: I believe these were based on measured values back then; yes.

MEMBER ABDEL-KHALIK: Are these numbers consistent with the historical data for valve leak rates, like feedwater check valve leak rates, MSIV leak rates?

MR. KIRSTEIN: I'm not quite sure on that.

MR. McCULLOUGH: Well, the system here we're--this would be the offgas, which is processing the air ejector discharge. I was just a little confused by the question regarding--

MEMBER ABDEL-KHALIK: Well, I'm just trying to find out where the numbers come from.

MR. McCULLOUGH: Oh, okay. I'm sorry.

MEMBER ABDEL-KHALIK: And whether they make any sense.

MEMBER ARMIJO: They're extremely conservative, I think. I think that's a design basis.

MR. KIRSTEIN: Yes, that's our design basis for--

MEMBER ARMIJO: You know, compared to your experience. Maybe that's what--you know, do these numbers mean anything--

MEMBER SHACK: This is their worst day.

MEMBER ARMIJO: Yes. Absolutely worst.

MR. KIRSTEIN: I mean, we went out to choose quite a conservative value for our design basis for the noble gas rates.

MR. KRESS: For the Iodine-131, does that include what's called the "iodine spike" when you go through transients?

MR. KIRSTEIN: I'll have to check on that.

MR. KRESS: Oh, it doesn't, because you only have to worry about that a few days and decays away.

MR. KIRSTEIN: Yes.

MR. KRESS: But I'm just wondering.

CHAIR CORRADINI: Okay.

MR. McCULLOUGH: Okay. Next would be the solid radwaste, section 11.4, using the same basic process as the existing BRWs, and we take what solid waste from the plant equipment, filters and demineralizers, put it in a waste container, high-integrity container, it's dewatered and dried to meet burial site criteria, or to a waste processor, sent to a waste processor.

MR. KRESS: This is low-level waste?

MR. McCULLOUGH: That's correct, sir. Waste streams are segregated, so we have the B waste in one tank, in certain tanks so that that waste, which right now can only go to Barnwell, is separated, and by doing that, we end up reducing the total amount of waste we generate, and the cost is much higher for B waste. So that's the primary reason for segregation.

And once again, the solid waste systems are designed to meet the Reg Guide 1.143, and also factor in ALARA, and use cameras and road operating equipment.

The next section is 11.5, process effluent, radiation monitoring sampling. Again, we're similar to existing boiler and water reactors, to use that to diagnose our liquid and process streams, and have initiation functions for areas where contamination could be a problem, where you'd have an airborne.

And we have safety-delayed monitors for the closure of drywell sumps, isolation, condenser isolation valve, and containment purge.

And we have radiation monitor sample points to monitor gases, or liquid and effluent process streams, and instrumentation that's compatible for anticipated operational occurrences and accident conditions. As you see on the screen, applicable references that were used to support the DCD and the design reforms and in summary, as you see on the screen, the number of--we have open and confirmatory items which we're working with the staff to close for the five different subsections of Chapter 11.

CHAIR CORRADINI: Just to make sure I understand, because think I do, I just want to keep on reminding myself. The confirmatory items will be captured in the ITACCs?

MS. CUBBAGE: I can address that, if you like. Amy Cubbage. Confirmatory items are items where the staff has agreed with GE's REI response and proposed revision to the DCD--

CHAIR CORRADINI: Which is yet to be seen.

MS. CUBBAGE: Which is yet to be seen. In some cases, it may have come in in DCD Rev 4, which of course was not addressed in the SCR that was sent to you. So we either received in DCD Rev 4 or we expect to see it in DCD Rev 5.

CHAIR CORRADINI: And so it may be a even more precise part of a design that then settles the issue or it might end up as an ITACC?

MS. CUBBAGE: No, nothing to do with ITACC. These are open--these were open issues with the design control document, that have been resolved and will be implemented in a future revision of the DCD.

CHAIR CORRADINI: Thanks.

MR. McCULLOUGH: With that, I'll turn it over to the--

MEMBER MAYNARD: I've got a couple questions.

MR. McCULLOUGH: I'm sorry.

MEMBER MAYNARD: Just on the overall layout of the radwaste facilities. I take it you have cranes and other items incorporated into that area, to are able to move things. Camera systems that be remotely operated. Crane type things. Is that--

MR. McCULLOUGH: That's correct.

MEMBER MAYNARD: And instrumentation. Since a number of these may be on mobile skids, what kind of provision's been made for getting information to the control room, considering that there may be various, different types of skids used and stuff?

MR. McCULLOUGH: Well, the mobile skids will interface with the, you know, permanent plan, DCIS, and so all alarms will have the potential to go to the control room. They'll be screened by a committee of human factors. SROs basically determine which alarms they really want to bring into the control room. It may be a common radwaste trouble alarm. There may be some items that are of higher importance, that they would have a direct alarm come into the control room, but that would be a result of human factors, review, as to what alarms you bring in.

MR. WHITE: There's also local alarms and monitors for some of the items in the local radwaste control room. There's a separate control room for that. So we have a capability for some of those items.

MEMBER MAYNARD: Is there any operation of the radwaste system required during the first--I think this plant's set up for like the first 72 hours with no operator action during an emergency, plant transient or whatever. But any operator action required in the radwaste facilities?

MR. WHITE: No.

MEMBER ARMIJO: Somewhere in your documents, in design control document, I read that this, the plant is being designed so that the radwaste will be limited to something like 10 percent, the lowest 10 percent of currently operating PWRs. That's the radwaste generation. I don't know if you're--first of all, tell me if that's correct.

But this system is at the end of the line.

What happens if the plant generates more radwaste? Is this system capable--you know, what's the capability of this system in that event?

MR. McCULLOUGH: Well, the capability of radwaste, it's a robust system, but it's designs to handle the, you know, the effluent, the maximum waste you would get during an outage, for example.

Anything over and above that, we would, you know, if we had a big leak in the plant, a huge leak, then there would be some special actions that would be required outside your normal process. Radwaste is designed to handle the most liquid we would see during outage conditions.

CHAIR CORRADINI: And that's about--I was looking from another one of our member's questions, that's about 100,000 liters a day. It says from your section, your table in Rev 4, about 100,000 liters a day. It was changed; went up. Does that sound right?

MR. McCULLOUGH: That's pretty close.

CHAIR CORRADINI: Okay. Other questions? Okay. Thank you very much. Will the staff come up. We'll hear about staff's evaluation.

You guys all set?

MS. JOHNSON: Yes.

CHAIR CORRADINI: Okay.

MS. JOHNSON: Good morning. My name's Andrea Johnson. I'm a project manager in NRO and new reactor licensing, and I have with me Jean-Claude Dehmel. We will be reviewing the safety evaluation of Chapter 11.

I just want to point out, I realize that you have Rev 4 with you and you may have some questions on it. But I just wanted to emphasize that the safety evaluation that was submitted to you and our presentation today is based on Rev 3, plus any of the RAIs that we have received response on from the applicant.

The review team consisted of myself as the lead PM. Our lead reviewers were Jean-Claude, Jai Lee, Chang Li and Hulbert Li.

Our presentations today will include another review of the applicable regulations, the RAI status summary, the technical topics of interest, John-Claude will go through, the open items, and significant COL action items. And then of course any of your comments or questions.

I'm not going to go through these in detail but this is basically a summary of the applicable regulations and review guides that were applied during the review.

RAI status summary. Initially, there were eighty-eight original RAIs. We have resolved 85, with three remaining open items, which will be discussed, in detail, a little bit later.

I'm going to hand it over to Jean-Claude now.

MR. DEHMEL: Thank you. Again, my name is Jean-Claude Dehmel. I'm a health physicist in the NRO's health physics branch. I was responsible for the evaluations of the effluent source terms and system performances, and associated with the liquid waste management system, the gaseous waste management system, the solid waste management system, and process radiation monitoring system.

Before I proceed, I would like to point out this was a--the review of Chapter 11 of DCD essentially involves a multidisciplinary effort, namely for Chapter 11.1, Jai Lee is the lead reviewer, for Chapters 11.2 and .4, Chang Li from the balance of plant branch, and I, share some review responsibilities on balance of plant system as well as some of the health physics topic.

Similarly, for Chapter 11.5, Hulbert Li and I shared responsibilities on instrumentation and the associated health physics instrumentation aspects.

With Section 11.1, on source terms, the topics of interest focus on the design basis for normal operation using the NCNS 18.1 standard and Reg Guide 1.112, the NS standard is used to establish typical long term concentrations, primary coolant and primary steam for BWR.

MR. KRESS: Reg Guide 11--

MR. DEHMEL: 112.

MR. KRESS: Is that the same as the ANS standard? It's just repeated in a reg guide? Are they consistent is what I meant to say?

MR. DEHMEL: Yes. The reg guide offers two methods to calculate the source term in primary coolant and primary steam. One is essentially simply the adoption by reference of the ANSI standard 18.1.

The other method that's offered is the use of BWR GALE code, and in this case it's documented in your Reg 0016.

The design basis source term is used for the design of plant equipment and shielding but is not used for reactor accident source terms or accident scenarios. That's addressed separately in Chapter 15, which will be addressed at some future time.

The design basis source term reflects 1 percent fuel defect corresponding to approximately

100,000 microcuries per second, noble gas's release rate after 30 minutes decay.

For Section 11.1, the topics we reviewed in RAI focused on a description inclusion of parameters using, deriving nucleotide concentration, primary coolant and steam, identification of normal and potential sources of effluents, and clarification on source terms for fission activation and corrosion product, including noble gases.

The staff confirmed the source terms and found the source terms acceptable. All RAIs were a satisfactory result, all RAIs are closed, and there are no COL action items.

With Section 11.2, on the liquid waste management system, topics of interest focused on equipment design for normal operation, anticipated operational occurrences, and features to process, collect and treat--sorry--and treat liquid processes and control effluent releases. I'm having a problem here.

The system design relies on a mobile radwaste subsystem connected to permanently-installed equipment, as was described earlier by GE staff.

We've identified the key SRP interfaces here, 9.3, 11.3, 11.4, 11.5, and 12.2.

MR. KRESS: The actual concentrations and potential doses, that would be in Chapter 12?

MR. DEHMEL: Yes. I will be making the corresponding presentation about that a little bit later this morning. That's correct.

The topics review and staff RAI focus on consistency of tank design basis against Reg 4.3 system flow pass, process streams, effluent discharges, basis for system performance, express the decontamination factor, DF, in treating liquid waste, scope of COL action items for mobile waste processing systems, and ITACC on mobile systems configuration, plant system interfaces and operation.

MEMBER ABDEL-KHALIK: Can we go back to the previous slide.

MR. DEHMEL: Sure.

MEMBER ABDEL-KHALIK: Where you say the equipment design is for normal operations and anticipated operational occurrences, have you verified that all the anticipated--the complete set of anticipated operational occurrences for this particular design have actually been analyzed?

MR. DEHMEL: Not a complete set. Basically, the evaluation considered whether or not the system is added to--does adequately contain a number of tanks, tanks of sufficient capacity, and that the processing rates of the systems are adequate to process the anticipated volume and radioactivity levels of the expected liquid effluents.

MEMBER ABDEL-KHALIK: So what is the meaning of this statement, then?

MR. DEHMEL: The meaning of the statement is that in addition to being able to processing wastes, with respect to what would you expect under normal operation conditions, that, for example, the Reg Guide 1.012 and NUREG-0016, acknowledges there may be some anticipated operational occurrences, some minor plant upset, that are not essentially in the context of Chapter 15 type of scenarios. So these are minor. For example, let's assume that there's a spill and all of a sudden you have additional liquid waste, or that there is a failure of a component, thereby generating some additional sources of radioactivity, or perhaps highly concentrated liquid waste on the drain system--

MEMBER ABDEL-KHALIK: So there is a specific definition of anticipated operational occurrences, which is different than what we normally call anticipated operational occurrences in this context?

MR. DEHMEL: I'm not sure I follow the question. I think what I tried to say is it differentiates anticipated operational occurrences from Chapter 15-like scenario accident analysis which is different. Those are addressed in Chapter 15; aren't addressed here.

MEMBER ABDEL-KHALIK: So these are different and-

MR. DEHMEL: These are different. These are essentially minor operational upset that all of a sudden results in a generation of an additional 10-, 20,000 gallons of liquid waste, or perhaps results in higher radioactivity levels because a filter failed or something happened to the ion exchange resin.

MEMBER ABDEL-KHALIK: So again the question remains: How do you define these anticipated operational occurrences in your context?

MR. DEHMEL: I have no specific definition, other than recognizing that the system is sized and that the demineralizer columns, and the capacity of the tanks, and the flow rate of the pumps, adequate enough to address those anticipated occurrences. There's no specific list of scenarios containing the application that actually describes this.

MEMBER ARMIJO: I guess a question is, kind of similar question that I have is what's a margin in this equipment? Is it capable--you know, if your normal operating capacity is one number, what are these systems sized for?

CHAIR CORRADINI: I guess I'd follow on--

MEMBER ARMIJO: 1.1, 1.5, 2? What's the design margin?

CHAIR CORRADINI: I think that's GE's--

MR. UPTON: Yes. Sam, can I--

CHAIR CORRADINI: Yes.

MR. UPTON: It's Hugh Upton with GEH. Let me address that. The normal radwaste system is designed to process radwaste for an eight hour shift, 40 hours a day. Okay, that gives us the 100,000 liters per second. I'm sorry. 40 hours a week. and in the event of an extreme, say an AOO, where you have to process further, we could go to three shifts, eight hours a day, processing 24 hours a day.

So that's the kind of margin that you have in the system.

MEMBER SHACK: Plus the fact that you're not normally operating with 1 percent defective fuel--

MR. UPTON: That's correct. Plus we're not--that's right.

CHAIR CORRADINI: So I guess that's another way of--I'm trying to get a handle on--is that a question from one of our other members, and I'm just reading. So from DCD 3 to 4, you went from some number up about 10 percent in terms of the total up throughput. But in normal operation with current BWRs, I'm curious how this scales.

Does it scale on thermal power? Does it scale strictly on thermal power and the fuel defect as your upper design limit, and then I'm curious what normal operation for a fleet of plants you normally get, what's your margin. I think it goes back to Sam's question.

Are you a factor of three away from margin because you normally operate 40 hours a week on eight hour shifts? Are you ten times--do you what I'm getting at?

MR. UPTON: I understand your question. What we'll have to do is get back to you with the specific numbers. But first, radwaste doesn't necessarily correlate to, one to one correlation with power.

CHAIR CORRADINI: Okay.

MR. UPTON: We think that we are very conservative in the design. We think that we have sufficient margin. We think that in the event of an AOO, we've got more than enough capacity to handle it. The exact numbers, though, I'll have to defer--

CHAIR CORRADINI: That's fine. I don't expect you to extemporaneously give the exact numbers. But I was back to Sam's question about--

MR. McCULLOUGH: It's roughly three is what I heard.

MR. UPTON: Dale, did you want to mention something?

MR. McCULLOUGH: Just speaking from experience, prior to joining GE, I was radwaste supervisor at Quad Cities for the last eight years, and when I saw the design of the ESBWR, I noticed it's very robust. We had two units, you know, at 912 megawatt electric, we had only one collection tank, and during power--we went to a power uprate, and we really didn't see liquid, in actuality, liquid amount, liquid process go up. It was essentially the same. In fact, the amount of liquid we process through radwaste has been decreasing over the years, just through the fact that we've improved the plant, it doesn't leak as much and we're a lot more conservative with what water goes to radwaste. People are a lot more cautious. So just the terms. And we would process, typically, over every year, 12 to 14 million gallons total, and 12 to 13 is what we--the equipment side, primarily your high conductivity waste. So I saw, you know, qualitatively, in the ESBWR, a lot of margin, it's very robust, compared to what I was used to operating with at the Quad Cities station.

MEMBER MAYNARD: I don't think we have the right people, necessarily, to answer some--there is a fairly clear delineation between watts and operating a normal expected operating occurrence as opposed to a design basis accident, and typically, there's one or two of those that really set the limits. Most of the others fall well below that.

So it's not unusual that they don't necessarily evaluate every single operating occurrence. And typically, it ends up being like a reactor scram, is one of those expected operating deals, and usually the limits are, you know, a reactor scram wit the most failed fuel that you're allowed to have for normal operation, which nobody really operates at.

But I don't think we have the right people to answer some of the--you know, what is the clear definition. But it is not just a guessing game as to what's an abnormal operating occurrence.

MEMBER ABDEL-KHALIK: My concern is terminology. The term anticipated operational occurrences means something specific in Chapter 15 space, and I just don't want these terms to be confused.

MEMBER MAYNARD: Well, to add some more confusion to it, they've actually changed that terminology and stuff, too, that's used in the regulations and reg guides too.

CHAIR CORRADINI: Are we ready to go on?

MR. DEHMEL: You know, just a point of clarification here. The DF is used to express the performance of treatment systems such as, for example, a DF of 100 for any exchange resins, a DF of 1 for filters, a DF of 1 for tritium, and a DF of 1 for diversion waste. Essentially this is the kind of information that's used to ultimately derive the source term, meaning the source term that goes out the stack or goes out through the discharge pipe.

The effluent monitoring system is tied to the COL action items identified in DCD Section 11.5. We're going to talk about this a bit later.

One RAI remains open on mobile ... processing system. This is RAI 11.2-16. It's on page 11.14 and 15 of the SCR. There are seven items associated with that. And the DCD identifies 12 COL action items. I see two COL action items. Next slide, please.

The Section 11.3 on the gaseous waste management, the topics of interest focus on equipment design for normal operation, and again anticipated operational occurrences in a context of Chapter 11, not 15, and features to process, collect and treat gaseous process stream and control effluent releases.

As opposed to a liquid waste management system, this portion of the system relies on permanently-installed equipment, and I have listed here the key SRP interfaces. Next slide, please.

Again the topics reviewed, and staff RAIs focused on the qualifications of the old gas system to withstand internal explosions, system design features and specification, basis for system performance, in this case holding time in treating gaseous waste, and scope of COL action in defining system performance and effluent monitoring. Again a holding time here is a surrogate for performance in retaining noble gases in a charcoal delay base.

For example, for xenon it's about 60 days, for argon it's about four days, and for--I'm sorry--for krypton it's four days and for argon is about one day.

One confirmatory item remains open on the COL holders QA program, and again the effluent monitoring portion of the system is tied to COL action item as discussed in DCB 11.05.

For the solid waste management system, the topics of interest focus on equipment design for normal operation and speedy operational occurrences, and features to process, collect, and treat solid and wet waste, and control effluent releases. The system relies on a mobile radwaste subsystem connected to permanently-installed equipment.

In this case, the evaluation also addressed in one operation program, the process control programs, identifies COL action item as was mentioned earlier, and we identified, here again, the subtle interfaces with the SRP.

So the topics reviewed and the staff RAI focus on the consistency of the design basis against Reg Guide 1.143, system flow path, process streams, licensing discharges, methods for processing large components, spent charcoal, scope of COL action items for mobile waste processing system, and ITACC on mobile system configuration, plant system interfaces and operation. Again, the effluent monitoring portion of this is tied to DCD Section 11.5 and will be addressed with my next section.

Two RAIs remain open, ITACC and DCD, one ITACC and a DCD scope mobile system, and Chapter 11.4 identifies twelve COL action items.

For the process radiation monitoring systems, the focus addresses the equipment design for normal operation, anticipated operational occurrences and features to characterize types and amounts of radioactivity in process streams and effluents, and control effluent releases.

The system design relies on a combination of a skid-mounted subsystem, currently-installed equipment. Again, this section focused on the operational program, three of them, mainly the outside dose calculation manual, the standard radiological effluent controls, and radiological environmental monitoring program, all as part of COL action items, and again we identify the same series of SRP interfaces.

For the process radiation monitoring system, the staff RAI focused on the design basis against SRP Section 7.5, 11.5, and Reg Guide 1.21 and 4.15. Reg Guide 1.21 addresses measurements and reporting effluent releases to the NRC, and Reg Guide

4.15 addresses quality assurance and control for radiation monitoring equipment.

The RAI focused on instrumentation systems, sample stream, and effluent discharge points. We also addressed and looked at automatic safety function isolation and termination of releases, and the scope of COL action items for instrumentation systems and operational programs, again tying this back to the operational program which have to be in place before fuel loading.

And there are 18 confirmatory items that remain open, and the DCD Chapter 11.5 identifies five COL action items.

MEMBER SHACK: Just out of curiosity, I would have guessed that the SRP section and the reg guides for these things haven't really changed very much. I'm just sort of wondering why so many RAIs were needed for something that I would have thought the guidance was pretty good for.

Were there changes that--I mean, were there significant changes in the guidance, that would indicate that, you know, there's a reason?

MR. DEHMEL: No, the--well, two things. One is the DCD was prepared against the 1981 version of the SRP, and some of the reg guides. The March

2007 version of the SRPs, Chapters 11.1 through 11.5, have been edited, but none of the fundamental changes, none of the final guidance and SRP criteria have changed. For example, we have provided some additional elaborations on the content of the operational program. We also make greater emphasis on the requirements of 10CFR, Part 20, 14.06, and so on/ The basic criteria, and SRP guidance and reference to existing regulations have not changed.

The issue with the RAI essentially, you know, addresses the staff's review, and finding out internal inconsistencies are now being crisp and clear about, you know, how aspect of Part 20 or Appendix I has been implemented in design, or how these things will be then carried over as COL action items. These are the kind of RAIs that were identified, not necessarily that the DCD, you know, is completely ignorant of the reg guides or the SRP.

It's just further clarification, further information, and also for the purpose of making sure that it was clearly understood that DCD addressed certain elements, and there was a delta, and a delta had to be addressed by the COL applicant.

MEMBER SHACK: Okay.

CHAIR CORRADINI: So I guess those

questions kind of are going where I was, and actually goes back to the design. So there's nothing about the radwaste--let me ask it differently. There's nothing about the--I'm looking for deltas. Is there a big design change in the radwaste systems from this and a current BWR, and what I heard from GE was, it was essentially these mobile skids to allow more flexibility than what you might need and what you then have can change as the plant operation continues.

But except for that, it's pretty much the same, same source terms, same all this. And so your answer to Bill was the delta here is not so much the difference in the SRP, it's just the level of detail you were able to look at, and how it addressed the regulations, needed some clarification.

MR. DEHMEL: That's correct.

CHAIR CORRADINI: Okay. So let me ask the broader question, maybe for GE, or you can start with it. So what is different on this part of the plans than a current plan, that you would focus on or look at carefully, to make sure something didn't -- if different, in a way that concerns you? Or is it essentially the same set of radwaste systems we see at current BWR?

MR. DEHMEL: It's different in a sense

that in light of the emphasis on 2014.06 and the concern about confirming that--or avoiding unmonitored on release, and unmonitored releases and uncontrolled releases to the environment. The focus here has been on looking at mobile systems and making sure that once you slip into the plant system, this mobile system, that by doing so you're not introducing potential paths on monitored and uncontrolled visas that would not be captured, for example, by one of the effluent radiation monitoring systems. That was one issue.

CHAIR CORRADINI: Can you help me. Say that again. I don't think I appreciate what you just say. I'm sorry.

MR. DEHMEL: Just imagine you have essentially a number of pieces of equipment supporting the operation of liquid wet waste. Right. So you have tents and pumps and so on. And then you have a discharge pump with a radiation monitor. In between, the utility would actually insert what is a skid-mounted system. But that skid-mounted system requires plant support, interfaces from compressed air, from water, and so on, and the idea is that if the plant, if the DCD had already included the radwaste system as part of the DCD design, we would be able to look at the design and confirm that perhaps, with the level of information that was provided, there was no opportunity, or at least the design considered opportunities and avoided conditions where there might be unmonitored, uncontrolled releases--

CHAIR CORRADINI: From somewhere in between where you plugged it in to where you're normally monitoring. Is that your point?

MR. DEHMEL: Right.

CHAIR CORRADINI: Okay.

MR. DEHMEL: So here, with mobile system, we don't have the opportunity, because as you can see in Rev 3, even in Rev 4 of the DCD, everything's identified as conceptual and the level of detail is not the same as you would expect for a permanently, you know, described system.

CHAIR CORRADINI: Got it.

MR. DEHMEL: So the focus was on these kind of interfaces and making sure that we were able to focus on this and that these would essentially be important COL action items, so that when the utility decided to select a system, that it will be a reminder that, oh, by the way, you know, in addition to confirming that the system met the performance requirement, for example, as a DF or as a holding time, you also had to be concerned about potential, essentially bypasses, so to speak, for radioactivity to be released to the environment without being monitored and then uncontrolled.

CHAIR CORRADINI: Right. Thank you. That helps. That helps.

MEMBER ARMIJO: Okay. So there's no new technology or novel application of old technology in this system, that the staff is worried about, that, you know, something that could fail and lead to safety consequences downstream?

MR. DEHMEL: No. The technology that's described in the system is fairly straightforward. Ion exchange resin, reverse osmosis, filtration, and so on. So really nothing, there is nothing unusual here with respect to, for example, introducing the second or third generation of operators, introducing the kind of waste processing techniques or systems or processes you would find, for example, in a hazardous waste area. You know, there's nothing of that here.

It's fairly straightforward, conventional type of equipment.

CHAIR CORRADINI: So just to pursue that point, there's one last step to go with what Sam's asking. In terms of the radiation monitoring or the instrument, is there anything there different? I mean, is there any more heavy reliance on digital instrumentation that might cause one to have a different sort of failure? You see what I'm--I'm just looking for differences.

MR. DEHMEL: Again, with respect to radiation monitoring, the selection and the deployment of radiation monitoring systems are fairly straightforward. You know, it's again sodium iodine detection.

CHAIR CORRADINI: Okay.

MR. DEHMEL: Ion chambers. You know, and so on. The interface with the overall I&C is obviously digital. You know, once the signal is out of the detector, then at that point it's digital, and I think somebody will be addressing the I&C section later on.

CHAIR CORRADINI: Okay; that's fine. But that's a connection, I guess another one of our members sent a note, worrying about the connection back to any sort of new instrumentation, to understand those implementations.

MEMBER MAYNARD: I think that the potential for new concerns is going to come at the COL stage when the skids are selected. It may be a conventional skid that everybody's familiar with, or it may be a brand new design that is a first of a kind, and that's where I think some of the real new issues may, are going to come up.

And I had a question for you on that. This process seems to be establishing the criteria, the design parameters and what the skid ultimately has to be able to do, but since the skids aren't there, at the COL stage, does the staff plan to review these skids? Or if the COL applicant comes in with a design that says it meets these requirements, do you have to review the specific skid?

MR. DEHMEL: Yes. Yes, we do, for two reasons, remember that, and we'll talk about the doses later on. But the effluent source term, out of a liquid waste management system, out of a gaseous waste management, that's what's used currently to assess doses to the outside receptor, compliance with Part 20, appendix B, effluent concentration.

So, right now, this whole thing hangs together because there is a conceptual system, but the key to it, in a way, at this stage we, the staff, we don't care whether or not the system is gold-plated or chrome-plated or blue in color. The key is really the performance of the system as expressed by decontamination factor and as expressed by retention time for the gaseous, you know, effluent, the out-gas system. That's the key. So there are doses, there are effluent source terms in the DCD. We looked at it and we found it acceptable, you know, pending some issues that we talk about later on.

But ultimately, when the applicant comes in, if they actually splice in a new system that's not described here, or state-of-the-art technology we have not seen yet, or something associated with a topical report that's been reviewed by the staff, all the doses, the sourcing -- we'll have to recalculate it, to make sure that again, the concentration and Part 20, Appendix B, are met, that in Appendix I, those objectives are also met.

And also you have to factor in that you have a site-specific situation where the assumptions are used for chi over q and d over q, and in plant and offsite dilution for liquid effluents, are essentially, would be site-specific and will be different than what's assumed in the DCD at this point.

So this aspect will have to be totally reevaluated at a COL stage.

CHAIR CORRADINI: Good.

MEMBER ARMIJO: I guess I don't understand the advantage of delaying this design, either to the applicant or to the utility. You know, if it's this important, why in the world isn't it just made--you know, whether it's skid-mounted or not skid-mounted, why isn't it more complete at the DCD stage?

MS. JOHNSON: I think Jean said in their presentation that they plan to actually make it a part of the design. In Rev 3, it was identified as conceptual, that they plan to--they're changing--

MEMBER ARMIJO: Okay. So we'll be seeing--

MS. JOHNSON: That's to be, to be sent to us. We haven't received that yet.

MEMBER ARMIJO: Okay. Thank you.

MS. CUBBAGE: Jim. I'd like Jim Kinsey to speak to that.

MR. KINSEY: Jim Kinsey from GE-Hitachi. We've continued dialogue with the staff in working through the closure of remaining open issues and one of those topics is around this issue of conceptual design, and we're moving down a path, now, of, in the next DCD revision, we're moving that conceptual language and providing a specific description of a design, with the understanding that, you know, five years down the road, a COL applicant may decide on a different or a newer technology and they deal with that through the departure process.

CHAIR CORRADINI: Okay. Other questions by the committee?

MR. DEHMEL: Here is two concluding slides. So at this stage, right now, we have three open RAI associated with the status of mobile system, whether or not they are within the scope or out of scope of the DCD, and the associated linkages with the COL action items on plant interfaces.

And the resolution of the open RAIs are expected to be closed in the context of DCD Rev 4 and Rev 5. So here we have a number of COL action items, and on the order of twenty at this count right now, but we expect that to change, and essentially the focus is again on plant and site-specific features, define the COL stage. The big ones obviously are the COL action items for mobile processing system and plant interfaces, and obviously the COL action items associated with the operational program which only the COL applicant can address.

And then the resolution of COL action items are expected to be completed in a context of DCD Rev 4 and Rev 5 updates. And that concludes my presentation, and if there are any questions?

MEMBER ABDEL-KHALIK: I'd like to go back to the source terms. You indicated that it's based on historical data with 1 percent fuel defect.

What is the fuel inventory in the ESBWR vis-a-vis the BWR-6?

MR. DEHMEL: The inventory is based on the ANSI/ANS standard 18.1, and there is one specific inventory for all the radionuclide, broken down in several categories, and then the adjustments that are made on a plant-specific basis are the amount of water in a reactor vessel, the steam flow rate, and--

MEMBER ABDEL-KHALIK: Let me just be a little more specific in my question, just so that--the statement is that this is based on a historical GE value with 1 percent fuel detect.

Has this been adjusted for the fact that the total fuel inventory in the ESBWR core may be quite different than the inventory in the BWR-6?

MR. DEHMEL: Yes. The adjustment effectively reflects the thermal power level. The major adjustment for the thermal power level.

MEMBER ARMIJO: There are a lot more fuel rods, though; a much bigger core. You have a lot more fuel rods. So if it's based on 1 percent defect, you'll have 1 percent of a bigger number.

MR. DEHMEL: I understand. But the only adjustment in the methodology addresses itself to the reactor, the thermal power.

MEMBER ABDEL-KHALIK: Would the licensee care to comment?

MEMBER ARMIJO: Yes. It doesn't make sense.

MR. KINSEY: Jim Kinsey from GE Hitachi. There's a brief description in the DCD, that recognizes that we're at an increased power level, but also recognizes that there are improved fuel designs which, you know, tend to mitigate release rates. So the output here, or the source term that was selected is associated with those factors.

There's a reference, that I'll look at it here, maybe we can get back to this after the break, but there's a reference that's associated with how the source term was specifically developed.

MEMBER ARMIJO: Yes, but, you know, you see where we're coming from.

MR. DEHMEL: Yes.

MEMBER ARMIJO: If 1 percent fuel defect is a criteria, it doesn't really--for the same thermal power, you just use a lot more fuel, the source term's going to be different. If you go through the arithmetic, I think it'll be different. But I'd like the staff to think about it.

MR. DEHMEL: You know, I understand your question. Yes. I understand your question.

MEMBER ARMIJO: Yes.

CHAIR CORRADINI: Any other questions?

[No response]

CHAIR CORRADINI: Okay. Thank you very much.

MR. JOHNSON: Thank you.

CHAIR CORRADINI: We plan to have an hour presentation by GE on Chapter 12. Maybe this is a good time for a break till quarter of.

[A recess was taken from 9:30 a.m. to 9:48 a.m.]

CHAIR CORRADINI: Okay. Let's get back together. So we will begin by talking about having GE talk about Chapter 12 of the ESBRW DCD.

MR. KINSEY: This is Jim Kinsey from GE Hitachi. If it's all right, we'd just like to take one moment, while it's fresh in our mind, and go back to one of the issues from Chapter 11 that was associated with anticipated operational occurrences. Frostie or Dale, if you want--just so that we don't leave that one on the table.

MS. WHITE: In terms of how we design the radwaste equipment in terms of volumes, which I think Jean-Claude did touch on, we did look at AOOs that do generate waste, not all of our AOOs would generate waste, and we took the limiting case, and basically back-calculate what those volumes would be.

So AOO, as we define it, and as you saw on our slide, is actually Chapter 15.

MEMBER ABDEL-KHALIK: So there is no inconsistency. These are anticipated operational occurrences as defined in Chapter 15?

MS. WHITE: Yes, sir, and we look at the limiting case, back-calculate with the volumes we need to process that waste.

MEMBER ABDEL-KHALIK: Now the question then remains: How do you know that all the anticipated operational occurrences have indeed been analyzed, or identified?

MS. WHITE: We have identified in DCD Rev 3, and currently in Rev 4, the limiting AOO cases, currently. We have identified them.

MR. KRESS: I think that's always a complete misstatement when--

MEMBER ABDEL-KHALIK: Right.

MR. KRESS: And I don't think you ever

know if you've gotten all of them, for a new design. It's a good question, though.

MEMBER ABDEL-KHALIK: I guess we'll just wait till we get to Chapter 15.

MEMBER ARMIJO: Yes.

MS. WHITE: We certainly can elaborate on that when we get to Chapter 15.

MEMBER ABDEL-KHALIK: Thank you.

MR. SHUAIBI: This is Mohammed from the staff. I guess one thing we can do is we can come back at the full committee and see if we could address better the question.

MS. WHITE: So with that, we're the same crew up here again as Chapter 11, since they're intertwined, and I'll turn it over to Erik Kirstein to address the Chapter 12.

MEMBER ABDEL-KHALIK: Excuse me, before we go--there was another question pertaining to Chapter 11 which related to the source term.

Is that issue that you will address later on today?

MR. KINSEY: We're gathering some information that we would expect would allow us to come back to that before the end of the day, or we'll at least touch on that issue.

MEMBER ABDEL-KHALIK: Thank you.

MR. KIRSTEIN: I'll be talking about DCD Chapter 12. In my presentation I'll be covering a brief overview of the Chapter 12 contents, the various design parameters associated with Chapter 12 in the CDC, applicable references and then a brief summary of the inventory items.

As an overview, the administrative program is to, along with the design, ensure that the occupational radiation exposure to personnel will be kept ALARA. I'll be discussing the various subsections of DCD Chapter 12, 12.1, discussing ALARA, 12.2 is radiation sources, radiation protection is 12.3, dose assessments 12.4, health physics is 12.5, and then we created DCD Section 12.6 to address the minimization of contamination and waste generation, which was set up to directly address the requirements of ANSI Part 20, 14.06.

The following, on this slide, and the next one, are the COL items in Chapter 12 of the DCD. As you can see, the demonstration of compliance with the following reg guides, 1.88, .8, and 8.10. Providing criteria, conditions under which operating procedures and techniques are employed to ensure exposures are ALARA, utilizing the guidance of NUREG-1736.

And I think we saw this in Chapter 11. Ensuring offsite doses for both liquid and airborne effluents in doses -- yes -- the doses for liquid and airborne effluents comply with the applicable subsections of 10CFR 50, Appendix I, 10CFR 20 Appendix B, and 10CFR 20, 1301 and 1302.

Following on with some more COL items, the procedures provided for operation and calibration of air and radiation monitors, and the placement of portable monitors. A detailed description of the operational radiation production program, health physics, equipment, instrumentation and facility, detailed descriptions, and lastly, a description of the similarly unportable instruments for measuring radio-iodine concentrations under accident conditions and then also the training and procedures of said instruments.

Here is a list of--the following are Chapter 12-applicable regulatory requirements as associated to design parameters.

DCD Section 12.1 discusses ensuring that occupational doses are ALARA. The general design considerations for ALARA exposures are obviously the minimization of time spent in radiation areas and minimization of the radiation levels. This is implemented through the equipment design and the facility layout design.

The design considerations for equipment for reduction of ALARA exposures are equipment accessibility, the facilitation of maintenance and equipment materials. In terms of facility layout design considerations, to maintain exposures ALARA, we consider the allocation of equipment, the need for performing service of equipment in lower radiation areas versus higher radiation fields, and also providing adequate space for removable or portable shielding during operational activities in the plant.

DCD Section 12.2 discusses the radiation sources. As a brief overview, the following here are a few examples of the radiation sources described in DCD 12.2. We have the core sources in the reactor vessel, flux and gamma spectra, various equipment and system sources like heat exchangers, radwaste tanks, etcetera.

As discussed earlier, the airborne effluent releases and the resulting doses, the offsite doses, in accordance with 10CFR 50, Appendix I, for both airborne and liquid effluents. And also in 12.2, we discuss the onsite airborne sources during normal operation and also during refueling.

DCD Section 12.3 discusses the radiation protection design. In this section we discuss the radiation zoning. The radiation zone maps are provided for normal operation and shutdown conditions.

The specific radiation shielding in areas is discussed. Ventilation systems. The area radiation monitoring and radioactivity monitoring instrumentation for normal anticipated operational occurrences and accident conditions.

The post-accident access requirements. The access and egress routes. The operator actions and control points. And also the radiation zone maps, utilizing the highest expected dose for post-accident conditions. These are based on NUREG-0737, the vital area access, vital meaning equipment and systems required or needed to be accessed in a post-accident environment.

In DCD Section 12.4 we discuss dose assessment. The highest-expected doses are provided, assessed for the following activities, one of them being the drywell dose, with some examples of drywell dose functions or calculations.

MSIV main steam isolation valve repair, the safety, early valve maintenance and testing. By-motion control rod work and maintenance, and in-service inspection. Reactor bits are provided for operations. For reactor pressure vessel, the access and reassembly, refueling operations and control rod drive, control rod drive, hydraulic control unit work.

The fuel building doses are provided for refueling activities also. Turbine building doses for overall of the turbine and condensate treatment.

Radwaste spilling doses for maintenance of equipment, handling of radwaste shipments and radwaste processing as well.

And lastly, work at power doses for health physics coverage, surveillance activities and minor equipment repair are discussed in DCD Section 12.4.

DCD Section 12.5. The majority of the section refers to the COL applicant action items, mainly because this section discusses operations more so than design. We do provide information, though, on the location of the healthy physics facilities in the service building, and then there are a couple of COL action items. The applicant will provide the description of the health physics equipment, facilities, and also a detailed description of the operational radiation protection program.

As I said earlier, DCD Section 12.6 was created to provide the compliance or discuss compliance with 10CFR 20, 1406. In this section we discuss the minimization of contamination through various design features. A few examples are the stainless steel-lined equipment and sumps. As we talked about in Chapter 11, the skid-mounted radwaste systems or mobile systems. A spent fuel pool has a liner and a leak detection system.

And middle concrete wall shield wall construction. Just to touch on that a little bit, the blocks we'll be using for temporary shielding are essentially concrete plugs surrounded by steel for ease of decontamination and to eliminate the possibility of leeching of contaminates in the concrete block walls.

MR. KRESS: Excuse me. This is just for comment. When I hear the word "minimize" I generally think of something versus another when you get to the minimum value, trading off one thing for another.

What is your tradeoff when you talk about minimizing, say, waste generation? Is it cost, or is this a minimum? Or based on what?

MS. WHITE: Well, certainly we look at cost, but also volume reduction, of course, is one thing we look at. And certainly minimizing contamination and classification of waste is a big issue. Obviously, there's a limited number of places you can send certain classifications of waste, and all of those were taken into account.

MR. KRESS: It's not your standard mathematical concept of minimization, though. It's just minimize, given what you can do and what your design looks like.

MR. McCULLOUGH: I would say a typical, one big advantage would be in the generation of tracked waste. In the drywell now, we don't have recirc pumps or recirc loops. There would be a huge decrease in the amount of DAW that would be generated, for the maintenance we won't have to do in the drywell.

MR. UPTON: If I might add a comment. This is Hugh Upton with GEH. Maybe we should change the word minimization to reduction. Contamination reduction.

MR. KRESS: Yes. That would, you know, for us that would help. But I guess it's been in use so long, we could probably adapt our concept along those lines.

MEMBER MAYNARD: I would think another major consideration would be on what the dose is. I mean, to minimize for whatever you want to call it, minimize contamination, if you end up getting more dose, trying to keep an area, try to eliminate it, a lot of these end up being dose--before you have the minimum dose.

MR. McCULLOUGH: I mean, that's a tradeoff in the current operating plans where you have, you want to minimize a square foot of contaminated area, but how much dose are you going to take deconning an area so that people can work in street clothes?

MR. KRESS: The reason I asked the question is I just wondered what the staff considers as acceptable minimization, or what they review and what they say. Okay, that's--

CHAIR CORRADINI: They'll have that answer when they--

MR. KRESS: Yes. We'll let the staff think about that.

CHAIR CORRADINI: So just to follow up on Tom's question but it was slightly different.

You list a whole bunch of things relative to design features. So I'm back to my question of delta. So if I went to Quad Cities--you're near Quad Cities, right?

MR. McCULLOUGH: I was. I was.

CHAIR CORRADINI: I thought so. Good.

It's nearby, so I can visualize that. If I went to Quad Cities and looked, are all the things you list here not typical of what you'd see in terms of a design? You don't see stainless steel-lined equipment, some skid--skid-mounted was talked about. And I'm sure you don't see concrete shield blocks with stainless steel linings.

So these are all new design features?

MR. McCULLOUGH: Yes.

CHAIR CORRADINI: Okay.

MR. KINSEY: I think, going on to--I think we touched on it a little bit, but the minimization or reduction of the generation of waste design features, of the liquid waste and solid waste management systems, the process streams and the segregation of waste allow for efficient processing and does attempt to reduce or minimize the total amount of waste coming out of the ESBWR.

You can see here, following are just some applicable references as it pertains to DCD Chapter 12. The standard review plan, the various regulatory guides, and NUREG documents.

There's some following slides. I won't go into those in much detail as the staff will address some of these, I think in greater detail, but you can see a list of the open and confirmatory items for the various DCD subsections.

That's about all I had.

MEMBER MAYNARD: I have a couple of questions on what lessons learned from current operating fleet might have been incorporated.

One is cameras. Does the design incorporate cameras, especially for health physics coverage, jobs and stuff? Remote monitoring stations?

MR. KINSEY: Yes.

MEMBER MAYNARD: Remote monitoring stations?

MR. KINSEY: Yes, sir.

MR. McCULLOUGH: And remotely-operated equipment as opposed to manual valves for equipment isolation. To keep operators out of the dose.

MEMBER MAYNARD: Okay. You might have just touched on this. My first part was for health physics coverage. The second part for operator rounds and stuff, areas that might be higher-dose areas. Cameras and stuff in those area to minimize time or number of times they have to actually go into an area?

MR. KINSEY: Correct.

CHAIR CORRADINI: Other questions?

MEMBER ARMIJO: In the choice of materials

I saw, I don't know if it was in this chapter or another one, that special efforts were done on picking materials that were low in cobalt. The inconels were very low percentage cobalt, even though they're high nickel, but their stainless steels were still--which you probably have a lot of--still has a pretty high cobalt content.

Is there anything in the GE plan or GEH plan to use very low cobalt, nickeled--stainless steels?

MS. WHITE: We'd like to defer that to our materials engineer who's here.

MEMBER ARMIJO: Okay.

MR. FREW: I'm Brian Frew. I'm the technical lead for materials. Yes. The answer to that question is we do plan to use lower controlled cobalt materials for the stainless steel parts of the systems.

MEMBER SHACK: Is that in the DCD somewhere, that commitment?

MEMBER ARMIJO: Yes. I didn't see it. It was sort of implied but I didn't actually see that that was going to happen. But it's in there some place, huh?

MR. FREW: In chapter four.

MEMBER ARMIJO: Okay. I haven't gone through that one.

CHAIR CORRADINI: Thank you. We'll learn more about that when we get to chapter four.

MEMBER MAYNARD: For materials, what about some of the components and stuff, such as valves, valve seats and stuff like that? That's another source for the stellite, for cobalt and stuff. So I take it that you're reducing it, you're trying to pick it in all those areas?

MR. UPTON: Mr. Maynard, this is Hugh Upton with GE. Let me address the stellite issue. The design of the ESBWR minimizes the use of stellite in valve seats for that very reason, to minimize radiation dose. I also wanted to address an earlier issue that you had about minimizing, well, radiation sources.

The basic design of ESBWR has fewer pumps, fewer valves, recirc loops, so the dose burden on the operator is significantly reduced. So just a comment.

MR. KRESS: When you replace stellite valves with some other kind of seat, is that a tradeoff between potential leak rate through the valve?

MR. UPTON: Leak rate and life expectancy.

MR. KRESS: You may have to change them off more often.

MR. UPTON: Yes; that's true.

MR. KRESS: Do you have a detailed design of valves that do not use stellite?

MR. UPTON: Yes. I think I'll defer that to Joel.

MR. MELITO: Good morning. I'm Joel Melito, the lead engineer for Chapter 5. The answer is, in general, yes, but not a per valve detailed design. But we would be insisting on eliminating cobalt from valves as a generic activity across all the valves, at least for the nuclear island, and we're working with our counterparts to make sure that happens throughout the plan.

MEMBER SHACK: I mean, you do cull out two valve seating materials, one stellite and the other a non-cobalt, and so you haven't actually decided which valves get what?

MR. MELITO: No. That decision has not been made.

MEMBER MAYNARD: And that's not always an easy decision. You don't want to do something where you end up with more dose, having to replace it more frequently. So it has to be looked at, but the philosophy needs to be reducing it.

CHAIR CORRADINI: Any other questions?

[No response]

CHAIR CORRADINI: Okay. Thank you.

The next team. Familiar faces. Go ahead, when you guys are ready.

MS. BERRIOS: Good morning. My name is Ilka Berrios. I'm a project manager in the GE Department of Reactor Licensing. Here we have Charlie Hinson and Jean-Claude Dehmel.

Today, we're going to be presenting a brief summary of the DCD application Chapter 12, which is radiation protection and we'd be happy to answer any questions from the committee at any time.

The team for this chapter was myself, project manager, and Charlie Hinson as the lead team reviewer and Jean-Claude Dehmel was a supporting team member. We are going to be presenting the applicable regulations that were used during the review and RAI status summary, SCR technical topics, the significant open items, significancy of all action items.

This has the guidance that we used during the review, which includes different criteria, federal regulations, regulatory guides, NUREGs and the standard review plan.

Status summary. We had a total of eighty, of RAIs since the beginning. Fifty-six of them are resolved. We just have twenty-four open items, the reviewers will be discussing now. So I'm going to leave you with Charlie Hinson.

MR. HINSON: Hello. My name is Charlie Hinson. I'm a senior health physicist in the HP Branch. Before I get started in the individual sections, I wanted to clarify any problems that you may have seen in the numbering system. The standard review plan is numbered 12.1, 12.2, and then 12.3/4 is combined. In the early versions, 12.3 and 12.4 were separate. And then 12.5. And in numbering the safety evaluation, 12.1 is the introduction, so suchly, 12.2 corresponds to 12.1 in the site review plan.

And the way that the DCD was structured is that instead of including radiation protection design and dose assessment in a single chapter, they broke it into two. So that's how it's numbered.

Okay. the first section in the SCR is Section 12.2, ensuring that occupational radiation exposures are ALARA. And in this chapter, the applicant described the policy and design considerations to ensure that ALARA would be featured in the design of the plant. And they also described some of the equipment design considerations that they've incorporated in the design for ALARA, including what we just talked about, low cobalt and nickel concentrations and components. Minimizing crud traps, shielding components from each other, separation of high and low components, equipment designed to facilitate maintenance, are some of the features.

The assorted design considerations that were described were easy access for component maintenance and certain components can be moved to lower-dose areas to work on and to repair.

Shielding between radioactive sources, such as pumps, separating sources in occupied areas, using labyrinth entrances to cubicles that have high-radiation zones, and ventilation flow form low to high concentrations.

Okay. Some of the staff RAIs in the first section focused on description of design features to minimize those during operation. The DCD originally had features to minimize dose during decommissioning, and we felt that they needed to describe more about how they would minimize dose during operation.

So we asked that type of question and they responded with material selection, flushing provisions on components for change-out cartridges, for seals on pumps, etcetera.

MR. KRESS: How do you know when you've got enough ALARA? Is that just a subjective judgment call on your part, based on the experience?

MR. HINSON: Yes; right. Exactly.

MR. KRESS: There's no real measure of ALARA.

MR. HINSON: No; exactly. Right. Reg Guide 8.8 has, you know, is full of ALARA features and we--

MR. KRESS: It's sort of you know it when you see it--

MR. HINSON: Right. And we look to make sure--we look at the collective dose and just how they've incorporated, and, you know, if we see areas that, based on experience at other plants, that, you know, are not being incorporated, we ask why not, and those questions.

MR. KRESS: Okay.

MR. HINSON: Okay. A second one of our open issues, our RAIs, was listing examples of ALARA facility layout features such as work done on equipment in low-dose areas, and centralized control panels. The third open item, that's still open, was whether they had sufficient shielding around the reactor vessel to permit access to the upper drywell during refueling operations.

And there's one open item remaining in SCR 12.2, and that has to do with burnup of fuel, and we're reviewing that. We have the answer, in house, and we're reviewing it now.

There are four COL action items in this section.

MEMBER ARMIJO: What's that fuel question, issue on burnup?

MR. HINSON: Okay. The issue was they based their shielding on 35 megawatt-days per metric ton, and I'm not the original reviewer on this, he asked this question, but in looking at the maximum fuel burnup, there were some fuel assemblies that were higher than 35 megawatt-days, and so we essentially asked GE to do an analysis to show the activity differences, and we also had a independent contractor evaluate the activity differences between 35 megawatt-days and higher burnups.

MEMBER POWERS: 35 megawatt-days.

CHAIR CORRADINI: Yes. He means 35 gigawatts.

MR. HINSON: Gigawatts. I'm sorry.

MR. KRESS: Just a procedural question on COL action items. How do you track those? Is there going to be a separate document that says these are all the COL action items, make sure you don't lose them?

MS. CUBBAGE: This is Amy Cubbage. Actually, the design control document, GE's application, will be the official listing of all the COL action items that any COL application would be required to address.

MR. KRESS: So you'll go back to every chapter and pull the active ones from that chapter and--

MS. CUBBAGE: Well, the design control document has them listed in a separate section of every chapter, and then there's a roll-up listing of all of them in Chapter 1 of the DCD. There's a table.

We, in our final safety evaluation, we will refer to each one of their COL action items, so if we--

MR. KRESS: There's a good chance you won't miss any of them.

MS. CUBBAGE: Well, if there's a COL action item that the staff believes needs to be added, we ask GE to add it to the DCD. We can't impose them in our SCR.

MR. HINSON: Yes. The COL action items in the first section have to deal with compliance with reg guides 8.8, 1.8 and 8.10 and also with reg guide 1.70. So those are the four that are being tracked.

Okay. And the second section under plant sources, we looked at contained sources, source terms for core and major radioactive systems. Jean-Claude looked at the airborne and liquid effluent source terms and doses. And I looked at the sources for airborne radioactivity on site.

MR. KRESS: Do you have a comment on the question about the 1 percent failed fuel versus scaling to reactor power for the leak rate into the RCS? You know, the question was it looked like the scaling to power and massive water didn't really address the percent of failed fuel, and there seemed to be an inconsistency in the statements there.

MR. DEHMEL: Let me clarify this and we'll provide you a more formal response later on. The way the BWR GALE code, matter of fact, the way the PWR-GALE code works, is that--and the ANSI standard--is that it's based on thermal power.

When we say 1 percent failed fuel, it doesn't mean that we're going out there and counting the number of fuel pins that leaked. It's the inventory of radioactivity into the core. So we don't care how many pins are leaking. The point here is that its total radioactivity inventory in the core. So we're not counting, we're not essentially comparing the number of fuel pins, the number of fuel assemblies, and the idea of wrapping up the source term according to thermal power, because the assumption is that thermal power is directly proportional to the amount of fuel, therefore, the amount of power, the amount of radioactive inventory would be there.

That's the way the ANSI standard is structured as well as the BWR GALE code.

MEMBER ARMIJO: Okay. So there's a misunderstanding of what 1 percent failed fuel means.

MR. DEHMEL: Well, yes, I think that--

MEMBER ARMIJO: You didn't say no, I misunderstood it.

MR. DEHMEL: Yes. One percent failed fuel, the thinking is that the first thing that comes to mind, well, how many assemblies you have, how many pins failed in each assembly, and you say, okay, then one percent of all that--it's the core, it's an inventory--it's one percent of the inventory. Now it's conceivable that you could have fuel pins leaking at different rates for the same amount of radioactivity. But you might have say a 100 fuel pins. In other cases you might have a 1000 fuel pins; right? Because depending on leak rates. In other words, the kind of damage or defects that you would experience in a fuel.

So it's one percent. Yes. The guidance and the reg guides refer to 1 percent failed fuel, .25 percent failed fuel. But really what is meant, it's the inventory in the core, not the number of fuel pins that is essentially accounted, assumed to fail.

MR. KRESS: Is that something that needs to be clarified in the guidance, do you think?

MR. DEHMEL: In light of the question, yes; maybe.

MR. KRESS: It would make sense, one percent of inventory. That makes a lot of sense.

MEMBER ARMIJO: Yes. That's independent.

CHAIR CORRADINI: Keep on going.

MR. HINSON: All right. The staff focused on the following RAIs in the plant source section. We looked at the effects of N-16 in steam system, on offsite doses. We looked at the location and physical description of major contained sources in the DCD, because although they describe the major source terms in the DCD, they didn't describe the physical location, the dimensions, and the material so we could do confirmatory shielding analyses. So we asked for that information.

And also we looked at the calculation of airborne concentrations in each of the buildings.

Okay. Jean-Claude is covering that.

MR. DEHMEL: Thanks, Charlie. Again my name is Jean-Claude Dehmel. I was responsible for the evaluation of the source terms and doses associated with the releases from the liquid waste management system and the gaseous waste management system.

The releases from the solid waste management system are captured and treated by the liquid waste management system and the solid waste management sys--and the gaseous waste management system. So there's no separate discussion in a DCD addressing the source term associated with the operation of a solid waste management system.

Also for the sake of brevity, I did not include in this slide the listing of the regulation or the regulatory guidance documents, given that they were identified in my earlier, prior presentation on Chapter 11.

But the focus here, on the doses here to outside dose receptor, is centered around Reg Guide 1.109, Reg Guide 1.112, and NUREG-0016 for the BWR GALE code, and obviously, I guess, the ANSI standard 18.1-1999.

So the focus here is on complying with a key regulation, namely 10CFR Part 20, Appendix B for effluent concentration, for both liquid and gaseous effluents. The doses under 10CFR Part 20, 1301 and 1302, and Appendix B, Appendix I, Part 50 Appendix I design objections.

So the topics reviewed in RAI focused on informed parameters forming the basis of the gaseous and liquid effluent source terms and doses to outside receptors. The staff requested the applicant to provide information with which to independently confirm the corresponding effluent source terms and offsite doses, clarify, provide the basis of specific input parameters, include in the DCD full descriptions of the approach and parameter used in deriving both gaseous and liquid effluent source terms.

At this time there are two open items. They are identified as 12.2-9 and 12.2-15, on pages 12-11 and 12-14 of the SCR, and also there are two COL action items that demonstrate compliance with Appendix

I, Section 2, design objectives, and the ALARA cost-benefit analysis for a COL application containing plant and site-specific design features. And that's all I have.

MR. HINSON: Okay. Thanks, Jean-Claude. Okay. The next section reviewed was Section 12.4, facility design features, and there's several parts to this.

The first was a description of the facility and equipment design features for maintaining exposure as ALARA, and one of the major features that reduces doses in the drywell of this design is the no recirc pipes or pumps in the round reactor vessel.

So that reduces--the applicant estimates that reduces the dose rates in containment, in the drywell, by roughly 50 percent.

Also as we've mentioned before, low cobalt alloy was used, stellite is minimized, and colmonoy is used in some valves to replace stellite. Pumps have quick change-out connections, etcetera.

Plant shielding design was another area we looked at. The shielding is based on accessibility and closure levels, and looking at the plant layout designs, we ensure that the radioactive source, high radioactive sources are separated from each other.

That components that are highly radioactive are located in separate cubicles, are separated by shielding, labyrinth entranceways to cubicles that have high radiation doses. and we just look at the access traffic paths to make sure that that makes sense, and that people can access components that require high maintenance fairly easily.

And the next section was the ventilation system to minimize personnel exposures. In this design, the maintain the airflow from areas of low to areas of high potential contamination and the HVAC equipment is located usually in low radiation areas to minimize the dose to people maintaining and changing filters out in these systems.

And we looked at the area radiation and airborne reductive monitor description, which gives the location of the area radiation monitors and describes the systems.

And finally we looked at post-accident access to vital areas, and we asked a couple RAIs on this to have the licensee describe their post-accident zones on the map and to identify all the vital systems that needed to be maintained following an accident, and we asked that they provide that these components and these areas can be accessed and serviced without exceeding GAC19 criteria, five rem for the entire mission dose, for each of these vital areas.

All right. Section 12.4 of the SE discusses facility design features. Our RAIs focused on dose areas in accessible areas near the inclined fuel transfer tube, and that is a tube where they transfer the fuel from the containment building, the reactor building to the fuel building, and it's kind of a slanted tube that goes through several levels.

There's two accessible areas of the tube to check it for maintaining the tube, which have access controls and shield blocks to control access.

What we were concerned about, whether there were any other accessible areas around this tube in the various levels, that if a fuel transfer assembly was being moved from the top to the bottom, whether that would create high dose rates to people, you know, working around these tubes. So we asked a couple questions on that.

We also had our contractor do some confirmatory shielding calculations of various portions around this fuel transfer tube. We also looked at--

MEMBER ARMIJO: Before you go too far, to what extent are all your dose rate conclusions dependent on the water chemistry used in the plant?

For example, zinc additions. Is that built into the dose rate? Is that an assumption, that zinc's going to be used in this plant, is going to be the reference water chemistry, and so that you'll know that your doses, what your sources--

MR. HINSON: Well, they're going to use hydrogen water chemistry, so I mean, that affects the N-16 levels greatly.

MEMBER ARMIJO: Right.

MR. HINSON: So they've essentially multiplied the ANSI source terms for N-16 by a factor of six to come up with a source term for the N-16.

MEMBER ARMIJO: Right.

MR. HINSON: And also noble metal chemistry, they've done analysis to see how that would affect--

MEMBER ARMIJO: Zinc's supposed to keep things like cobalt in the core.

MR. UPTON: Sam, this is Hugh Upton. Let me add a clarifying comment about the design.

The current standard plant for ESBWR does not include zinc injection. We have made a determination that we don't think it's necessary. It was primarily in operating fleet to reduce the doses for the operator for maintenance around recirc pumps and recirc loops. We do have hydrogen water chemistry. We have designed the shielding in the turbine building to account for the additional N-16 coming from hydrogen water chemistry.

MEMBER SHACK: Now is that noble metal, or you're ready to handle a full hydrogen water chemistry and the shine from that?

MR. HINSON: We are able to handle a full shine from hydrogen water chemistry.

MEMBER ARMIJO: But there's no need for the zinc injection.

MR. HINSON: That's correct. There's no need for zinc injection. We have no recirc loops.

MEMBER ARMIJO: That's good.

MR. HINSON: One of our other RAIs focused on post-accident radiation zone drawings with vital areas and mission doses, like I mentioned before.

There are nine open items in this section and three COL action items.

The next section deals with dose assessment. The applicant described the dose-reducing measures and design modifications incorporated in this design to minimize doses, and then they came up with the resulting projected exposures to the plant, and for this design, they estimated roughly 60.4 person-rem per year, which is less than half of the current BWR operating exposure based on 2006 data.

MEMBER ABDEL-KHALIK: And how was this estimate made?

MR. HINSON: Well, they--in fact, one of our RAIs bases asked them to perform a Reg 8.19 dose assessment cause they did a dose assessment based on various areas of the plant and the maintenance jobs that had to be done in those areas and the man-hours, and they based it on some of their design.

So they give a breakdown by major components, how much hours a year they would take, how many persons would need to do this work, and the average dose rates, and they came up with that.

But we asked them to provide a analysis based on Reg Guide 19 which breaks it down by job function and work function, and they haven't responded to that yet, and want to see if that, you know, results in any different--because some of the RAIs that I'll discuss in the next slide, it seemed like the total man-hours were rather low for some of the functions in their analysis, and so we wanted to have them look at it again.

CHAIR CORRADINI: Can I ask a question about that. So if I understood it, their estimates were very low compared to historical, operational experience?

MR. HINSON: Right.

CHAIR CORRADINI: And the reasoning there was a redesign of the equipment, such that--or a redesign of how the personnel are used with the equipment that is there, and therefore they would be able to achieve those levels?

MR. HINSON: Right, yes, because like I said, there are no recirc loops in the drywell, and that's a major source of radiation for people during outages. And so by limiting those and if you assume that the dose rate drops by a factor of 50 percent, then that would, you know, knock a big chunk off, right there, and then there are lots of other design features that they describe in this chapter, that are novel ways to do things.

Robot/remote maintenance of areas, removing certain components to lower dose rate areas.

CHAIR CORRADINI: So I guess maybe I asked my--so you've explained what I was curious about but let me ask it differently. The current levels of occupational limits are not excessive. Because of

ALARA you want to reduce it.

MR. HINSON: Right; right.

CHAIR CORRADINI: Okay. So is there something about the new methods that may cause a concern in a different manner, that is, to achieve these lower levels, is there something about the design or the way the personnel are used, that caused you to--

MR. HINSON: Well, I think, like I said, the way that they did the analysis to come up with the number was somewhat different than we usually review based on Reg Guide 8.19. In looking--

CHAIR CORRADINI: That's more how they do the arithmetic.

MR. HINSON: Yes; right.

CHAIR CORRADINI: I'm curious, is there a procedure that they're starting to--is there something about the design or the procedure, of how the personnel would be used, that would cause you concern?

MR. HINSON: Well, like I said, we looked at some of the dose rates in the radwaste building, for instance, and they looked rather low based on current experience, and then we looked at the total number of person-hours that they estimated for the plant and that seemed also rather low.

You know, the design features that they've incorporated, and the resulting reduction in doses looked like, you know, they could--that the dose estimate was not really that far off, because if you look at the AP-1000 dose estimates there, it's a PWR but that's around seventy, which is not much lower than the current--

CHAIR CORRADINI: Seventy?

MR. HINSON: Person-rems a year.

Now PWRs have always been--roughly half of what PWRs are. And so the PWR design seemed to be, you know, still lower than the current generation. The BWR doses have really been dropping considerably. They've always been roughly twice as high as PWR doses but in the last three or four years, they've dropped considerably and they're, you know, catching up with the PWRs.

So I think, you know, it's not unreasonable to see that they could, you know, give an estimate of 60 rem. But like I said, we just want to look at the analysis a little bit more and ensure that those numbers--

MR. UPTON: Gentlemen, can I add a comment here. From the design standpoint, one of the reasons the doses are reduced has to do with the amount of equipment that we've actually eliminated from radiative areas like the containment.

We've got about 25 percent less pumps, 25 percent less valves that have to be maintained, and we've eliminated about 13 systems.

So the plant itself is significantly simplified, which means that the operator dose burden is much less, it's less to maintain, so that's one of the reasons that we're seeing that the levels are reduced.

CHAIR CORRADINI: Thank you for clarifying. So let me ask the question again to you. So that's all good, but is it always, always good, or is there something to the negative that's going to cause more operational, more need for operational maintenance that could up the exposure?

That is, with any sort of new design there's clear advantages but there's always another side of the ledger.

MR. UPTON: Based on the design that we currently have, we see no down side. In other words, we've reduced equipment, reduced systems, reduced the maintenance required on those systems. So the dose rate's going to be reduced.

MEMBER ABDEL-KHALIK: Can I just ask a

specific question about, just as an example, so we would know what design features have been made to reduce those. One of the things identified on your list, that gives the highest dose is the drywell dose for MSIV repair and SRV maintenance.

What specific design features have you incorporated to reduce that dose?

MR. UPTON: There are several design features. First of all, the plant itself, we don't anticipate that the SRVs are going to be cycled during normal operation, so we think that the maintenance on the valves will be much less.

During normal AOOs, we don't anticipate lifting any SRVs. For the main steam isolation valves, we have a maintenance room right off the main steam tunnel in a low radiation area that allows us to do valve maintenance. So those are a couple of the features that we've put into the design.

MEMBER ABDEL-KHALIK: Thank you.

MR. HINSON: I was just going to say that I was around in the '70s when we reviewed some of the current generation plants, and back then our standard for BWRs and PWRs was roughly 500 person-rem a year, and, you know, after TMI went up, but it's been dropping consistently since then, it's kind of plateauing out now but it's still slowly, you know, dropping, and so I think, you know, this design is considerably better than this. We're still looking at some of the analysis and haven't come up with a final number but--

CHAIR CORRADINI: Okay. Thank you. That helps.

MR. HINSON: Okay. Like I said before, some of the other questions I looked at were the justification of a low average dose rate for the radwaste activities and the total apparent low estimate in person-hours.

There are three open items in this chapter, which are the three that are pointed out here in those COL actio items.

Okay. Section 12.6 of the SE deals with the operational radiation protection program. Like we said before, this is pretty much entirely a COL action item. NEI has come up with a template to address Section 12.5 of the center review plan which is this section. And this template addresses organization, equipment, instrumentation and facilities, and procedures.

And we worked two years ago, quite, several months with NEI to come up with this template, it's been qualified, and and essentially GE is committed as a COL item to, you know, to use this template.

Okay. The RAIs in this section focused on description of radioactive sources to be used in the shielded rooms and health physics area, and also layout drawings of the health physics facilities in the service building.

Like I said, GE provided roughly a page to address this and so we had two RAIs on their page, and then the balance is going to be addressed by the COL applicant.

And three COL items are essentially the description of the operational radiation protection program.

Okay. T his last section is not, doesn't have a corresponding Standard Review Plan section. This was based on addressing the requirements of 20.14-06. This requirement was put into the regulations several years ago. There's no really reg guidance associated with it as yet. We have a draft reg guide that industry is going and staff has been working on for the last eight months or so. It's out in comment form right now and NEI has considerable comments on how you describe the minimization, or reduction, as you want to say, of contamination to facilitate decommissioning, and minimization of radiative waste generation. Those are the two pieces of this regulation that have to be addressed by all the applicants.

And like Mr. Kress said about how do you judge if there's enough ALARA, this is kind of the same. How do you, you know, know when to stop when you're talking about minimization of contamination?

So because staff is still looking at the comments from industry on how this reg guide is going to finally look, you know, we've kept these open issues for 14.06 until we can come up with a final reg guide and decide how we evaluate how many minimization features are enough.

Okay. The staff REIs. Back at the end of last year, before the reg guide was in existence, there was a NUREG CR-3587 that talked about decontamination facilitation.

Okay. The title is Identification and Evaluation of Facility Techniques for Decommissioning of Light Water Reactors.

And the staff asked an RAI how they complied with a certain section of that new reg since we didn't have a reg guide in place at the time.

So that's one of the open items. We asked them to provide features to minimize generation of radwaste during decommissioning versus during operation, and we also asked them to describe their features to minimize leakage from reaching groundwater, which is another important piece in this regulation.

There are three open items, which are these three that you see here. No COL action items in this section.

Okay. The significant SCR Chapter 12 open RAI items like I've talked about have to do with provision of post-accident radiation zones, clarifying their dose assessment, and also saying how they comply with 20.14-06. And resolution of these open items is expected in a context of Rev 4, Rev 5 in the DCD.

And significant COL items in Chapter 12 are description of the operational radiation protection program, like we said, including organization, equipment, instrumentation and facilities, and description of the radiation protection procedures.

Also we've asked for a description and location and calibration of airborne radioactivity monitors and description of access to control to Very

High Radiation Areas, which are areas greater than 500 r per hour. I'll open it up to questions.

CHAIR CORRADINI: Questions by the committee?

MEMBER ABDEL-KHALIK: What are the access control measures for Very High Radiation Areas?

MR. UPTON: Typically, it's administratively controlled. They're locked. The only access is by key. I mean that's typically how we've designed it in the plant. So there's no access during normal operation or even during an outage, without some control procedure.

MEMBER ABDEL-KHALIK: And that is explicitly specified in the DCD?

MR. UPTON: Erik, I defer to you. I defer to Frostie.

MS. WHITE: Frostie White. I've actually been involved with many decommissioning plants, so I've a lot of experience here. Typically, for contamination areas like that, and high-dose areas, we have lock and key and operational programs. And your high rad areas are fine, your tech specs as well, and you have to abide by those. So you have usually an operational program that addresses strictly high rad areas and access thereof.

MR. UPTON: In the DCD, as part of the rad zone maps, we define those areas that'll have to be under lock and key.

MS. WHITE: It's also a COL item in the CDC, that they provide a listing of those areas and a program to address that.

CHAIR CORRADINI: Other questions?

[No response]

CHAIR CORRADINI: Thank you very much.

MR. KINSEY: Excuse me. Just a point of clarification. Jim Kinsey from GE Hitachi. We had a little bit more discussion during this Chapter 12 session on the gas release rate and, you know, the clarification of 1 percent fuel failure.

Does the subcommittee have further questions on that topic? I just wanted to make sure we understand the status of that question, so we can work through--

CHAIR CORRADINI: In terms of the subcommittee members, are they satisfied now?

MEMBER MAYNARD: I am.

MR. KRESS: Well, we were kind of told it really meant one percent of inventory but it can't mean that.

MEMBER ARMIJO: That's too much.

MR. KRESS: It can mean one percent of some fraction of the inven--I don't know how you get to the actual value yet.

MEMBER ARMIJO: I know what it doesn't mean. I don't know what it means.

MR. KRESS: I mean, one percent of an inventory's a lot.

MEMBER ARMIJO: I know.

MEMBER ABDEL-KHALIK: I think it would be a good idea to clarify that.

MR. KRESS: Yes. I think that needs clarifying.

MEMBER ARMIJO: Maybe the staff could provide that. You know, what does it mean?

MEMBER POWERS: Roughly 7 megacuries. It's a little bit--

MEMBER ARMIJO: I know it can't handle that.

MR. KRESS: One percent of the gap inventory maybe?

MEMBER ARMIJO: Probably. Noble gases.

CHAIR CORRADINI: I would expect it'd have to be that.

Other questions by the subcommittee?

[No response]

CHAIR CORRADINI: Okay. So we're in an interesting situation. We're ahead of schedule.

I've been informed by the "powers that be, which aren't us, that this is a FACE-based subcommittee meeting, so we are not allowed to start in different so--but nobody told me we can't define what lunch is.

So my suggestion is that we take lunch now and begin at 12:15. I've been told that that's not allowed. We will try to fix that next time. I apologize to GE.

[Whereupon, a luncheon recess was taken at 10:45 a.m.]

A-F-T-E-R-N-O-O-N S-E-S-S-I-O-N

12:16 p.m.

CHAIR CORRADINI: So why don't we come to order and begin our afternoon session talking about Chapter 5 of the ESBWR DCD and Staff's evaluation.

So which one of you young men are going to start this off?

MR. WAAL: Right here.

CHAIR CORRADINI: Okay.

MR. WAAL: Good afternoon.

My name is Jeffrey Waal, I'm with the Regulatory Affairs GE-Hitachi. And I am the lead licensing engineer for Chapter 5 reactor coolant system and connected systems.

I'd like to introduce Mr. Jerry Deaver, who is the Nuclear Island technical lead, and he'll be doing most of the talking on this Chapter. And he'll be supported by Mr. Joel Melito, who is the Chapter 5 chapter engineer. And by Mr. Brian Frew, who is the technical lead materials.

Mr. Deaver.

MR. DEAVER: Okay. Yes. I'd like to give a summary of Chapter 5. Basically Chapter 5 is the reactor coolant system and connected systems.

What I'll present is the overview of

Chapter 5 initially and then get into the descriptions of each of the sections, and then followed by a brief summary at the end.

What I'm going to try to focus on is basically the changes that are different about ESBWR as compared to prior BWRs so that you'll see what has changed as opposed to what's standard and we've kept the same.

Chapter 5 the reactor coolant system basically involves all the systems that either transport fluid in or out of the reactor vessel and core region. And a bigger population or definition is what we call the reactor coolant pressure boundary, which basically is all the systems that are in containment that have high pressure and this boundary goes out to the second isolation valve on the containment. So it includes these valves and also the safety relief valves and depressurization valves that we have in the system.

In Chapter 5.1 that's basically a summary section that provides a summary for the entire chapter. What I'd like to do is go through three systems that are identified in this sections.

The first one is the nuclear boiler system. The nuclear boiler system for ESBWR contains the main steamline, and we have four of those, and also two feedwater systems or feedwater trains that deliver water into the vessel. Steam of course existing the vessel. And we have the traditional SRVs and safety values associated with the steamline.

The thing that's different about nuclear boiler system is the addition of the depressurization valves which are not on this mainsteam line, but they're connected with the IC system. The line that exists the vessel to the IC system.

So this is a very standard, typical system that we've had for BWRs.

What you'll notice on this figure is that we don't have any component or nozzles or systems that are below core, core being in this region here. And all these dotted systems are ones that are other attaching systems but are not part of the nuclear boiler system itself.

And we have the typical arrangement where we exit below from the safety relief valves down to the suppression pool.

CHAIR CORRADINI: So I can just --

MR. DEAVER: Yes.

CHAIR CORRADINI: -- to repeat what you said so I get the definition right. I was rereading the definition here as you had stated. Outer most containment isolation valves, second of two valves normally closed. And then the SRVs and the DPVs. So that essentially ends the boundary of the system as you defined it, right, on the left here?

MR. DEAVER: Yes. This is what represents the containment boundary.

CHAIR CORRADINI: Right.

MR. DEAVER: But it normally includes everything that penetrates into the containment of all the systems. This happened to be just a nuclear boiler system, though.

MEMBER ABDEL-KHALIK: Now eight of the SRVs that are designated as ADS SRVs, is there any logic as to the location of these? I mean, I was looking at the diagram. It didn't quite see any logic as to which ones are designated as ADS.

MR. DEAVER: Well, later one we have the actual diagram that shows the pattern of the SRVs and RVs. Can we look at that when we get to it?

MEMBER ABDEL-KHALIK: Sure.

MR. DEAVER: Okay. Thank you.

MEMBER MAYNARD: And I'm not sure I understood your point of what's not below the core versus -- I mean it looks like you have penetrations below where the core --

MR. DEAVER: Well, we have a normal bottom head penetrations, which are mainly the drain lines and CRD in-core penetrations. But as far as major nozzles like we've had before --

MEMBER MAYNARD: Major? Okay.

MR. DEAVER: -- the recirc system is not.

CHAIR CORRADINI: Can you just hold one second? They need to electrify you so that we can capture your words of wisdom.

So to get back to Otto's -- I guess Otto actually was thinking -- had the same thing. So except for the stuff at the bottom of the head, which are drain lines, clean up, et cetera --

MR. DEAVER: Right.

CHAIR CORRADINI: -- all the other penetrations into the vessel which are not part of the RCS --

MR. DEAVER: Yes.

CHAIR CORRADINI: -- but of course are part of the safety systems are above core level, is that correct?

MR. DEAVER: Yes, that's correct.

CHAIR CORRADINI: Okay.

MR. DEAVER: The next one is the iso-condenser system. Early reactors had an iso-condenser system but we haven't typically had that on a more current vessel. So this reactor -- this is at least a fairly new system that we're reintroducing in the ESBWR.

In this system we have a steamline that goes from the upper area of the reactor vessel, comes into the condenser which is in a pool of water outside of containment. And then we have a return line that comes back to the vessel.

This is a passive system. The main components are the condenser unit itself, which basically condenses steam in the event that the system is open. All the valves typically are open, but the main valves here are these two in parallel which are diverse valves. And once one of these are open, then the entire system is opened. And then any accumulated water in the system goes into the vessel and well as steam begins to condense in the condensers in the top part here.

CHAIR CORRADINI: So this is kind of like a test for us. But then to initiate this, as you said, all the incoming valves are opened. But to start it you open the out going valves through this new addition, which is the tank, and then drain that back into the system?

MR. DEAVER: Yes.

MEMBER ABDEL-KHALIK: So what happens on a loss of nitrogen during operation?

MR. DEAVER: On these nitrogen operated valves?

MEMBER ABDEL-KHALIK: Right. These valves would fail open, the top one at least would fail open.

MR. DEAVER: Yes.

MEMBER ABDEL-KHALIK: The other valve in the line, which is F004 fails to open so that entire inventory of the ICS would drain into the vessel during operation. Has that transient been evaluated?

MR. DEAVER: You know, Joe?

MR. MELITO: Yes. Actually it's a subset of the cold water injection transients that can occur. And so it's not the bounding event as such.

MEMBER ABDEL-KHALIK: So you have actually confirmed that that despite the large water inventory that you have in here --

MR. MELITO: Yes.

MEMBER ABDEL-KHALIK: -- that this is bounded by other like loss of feedwater transients?

MR. MELITO: Yes. Yes. Exactly.

MEMBER ABDEL-KHALIK: It is bounded?

Okay.

MR. DEAVER: This also shows the connection here of the system for the steamline with respect to the DPDs that are on that line also. Okay.

Any other questions?

MEMBER ARMIJO: What makes that system completely passive?

MR. DEAVER: Well, the fact --

MEMBER ARMIJO: I mean, do you have to activate a valve or does something --

MR. DEAVER: Yes. This system if you have a containment isolation event, these valves would open and would automatically start the IC system in operation. And the whole purpose of the IC system is to absorb heat, you know, from the reactor core region to avoid actuation of the SRVs and SVs in the system.

So based on the analysis of the anticipated operational events they show that there's really no event where an SRV would actually actuate, at least that's the expectation. And so this is system is what basically prevents those actuations from the SRVs.

MEMBER ABDEL-KHALIK: Now if go back to that loss of nitrogen transient, where do we see this evaluation. Is it in Chapter 15?

MR. DEAVER: It would be in Chapter 15, yes. Yes. And there was a question earlier about passive. Basically of the exit of this system, which is the steam, is at the upper elevation where the steamline is and the return is lower. So we have an elevation difference that facilitates the natural circulation in the system.

MEMBER ARMIJO: Some valves have to open for that steam to get up there.

MR. DEAVER: Well, it's an open system.

MEMBER ARMIJO: All the time?

MR. DEAVER: All the time.

MEMBER ARMIJO: Okay.

CHAIR CORRADINI: Just to follow up with Sam's question, I know that you guys explained this to us so I should remember this, but I'm sorry. But the line that's showing that's coming back down to main steamline is a purge line to continually remove noncondensibles even in the full operation state? The line that you were --

MR. DEAVER: Oh, this is actually the steamline.

CHAIR CORRADINI: That's the entrance.

MR. DEAVER: Right.

CHAIR CORRADINI: But then there was

another line coming down this one here that you're pointing. That's coming down.

MR. DEAVER: Yes. This one here is--

CHAIR CORRADINI: That's always open?

MR. DEAVER: That's open, yes.

CHAIR CORRADINI: And that's continually at a low flow rate essentially purging non-condensible buildup, have I got this correct?

MR. DEAVER: Yes.

CHAIR CORRADINI: Okay.

MR. DEAVER: Okay?

MEMBER ABDEL-KHALIK: So it's 15.4.2 -- 15.2.4.1 that transient gets analyzed. Thank you.

MR. DEAVER: And going to the third system, it's the reactor water cleanup and cooling system. What we've done in ESBWR is actually combined what used to be the RHR system and the reactor water cleanup into one system. And basically all the operations are the same as those two prior systems. What we've done is, you know, we're more efficient as far as the amount of piping and components associated with this system.

In the normal cleanup mode we have a low capacity pump operating which introduces one percent of feedwater flow to do the cleanup function. And then for shutdown purposes we would have both trains operating and we would have a higher capacity pump which would circulate more water, 7½ percent of feedwater that would aid the shutdown of the plan

CHAIR CORRADINI: So you're taking one percent of the flow, or some fraction of a percent of the flow --

MR. DEAVER: Yes.

CHAIR CORRADINI: -- and cleaning it up and then reinjecting with this system?

MR. DEAVER: Yes. It goes back through the feedwater system. It interjects in the feedwater --

CHAIR CORRADINI: Oh, I see. Okay. Thank you.

MEMBER ARMIJO: I guess I misread. I thought you had increased the capacity of the cleanup system and you were running at two percent all the time. You're saying it's one percent?

MR. DEAVER: With one train it's one percent.

MEMBER ARMIJO: And that's what you--

MR. DEAVER: We could operate with two percent by running both trains.

MEMBER ARMIJO: But that's not the --

MR. DEAVER: Normal mode should be with

one percent. We expect a clean system and, you know, we suspect that one percent would be adequate.

MEMBER ARMIJO: So the second train was really added to do the shutdown cooling?

MR. DEAVER: Shutdown cooling part, yes. Okay.

MEMBER ARMIJO: If only one train was operational, could you cool the system without any other active or passive system?

MR. DEAVER: Yes. It just would take longer.

Okay. 5.2 covers a number of areas. Basically it covers codes in code cases, reactor overpressure protection, the RCPB materials, preservice/in-service inspection and RCPB leakage detection.

With regard to codes and codes faces, we used the standard ASME code for design and fabrication of opponents. And the code cases that we're specifying are ones that have been approved by the NRC at this stage. We have some that are in process, but at this point we're only using basically code cases that have been approved.

MEMBER ARMIJO: Yes. I noticed that you're referencing for the containment internal structures, a new material that will require a new code case.

MR. DEAVER: Yes.

MEMBER ARMIJO: That's A709 HPS 70W, I don't know what that means, but what are the benefits or why are you using that as opposed to a -- you know, something that you have experience with.

MR. UPTON: Jerry, let me take a crack at that?

MR. DEAVER: Okay.

MR. UPTON: That's high strength steel. We're using it inside the primary containment because of the stress-allowables with that steel. We have applied for a code case. The code case is in process. I'm not sure I know exactly where it stands right now. But we are proceeding with that code case.

MEMBER ARMIJO: Okay. Well, you know, will this material be exposed to the coolant environment at all or --

MR. UPTON: No, no. It's strictly inside containment.

MEMBER ARMIJO: Okay. Just material properties, mechanical property.

MR. DEAVER: Okay. I'll proceed into the overpressure protection part. This is another diagram that basically shows the containment boundary, which is the dotted line. And it shows the primary system here with the main steamline, SRV and the safety relief valves and DPVs.

In this system the SRVs are setup with setpoints in the 1250 psi range. And they would be the first ones that would actuate. And those have discharge lines that go to the compression pool.

These valves can be manually actuated or can be spring operated. So they have more functions that can be used.

The safety valves are only spring actuated valves, and they're set at a higher pressure at 1270 psi.

DPVs are valves that are not pressure actuated. They're part of the ADS system so they actuate on other signals, such as low water level or containment isolate; other events associated with the ADS system.

MEMBER ABDEL-KHALIK: Now the manual actuation of these SRVs is what? Is there an electrical actuator inside the valves that's actuated from the control room or what?

MR. DEAVER: Yes.

MEMBER ABDEL-KHALIK: In addition to the

mechanical normal spring loaded actuation?

MR. DEAVER: The spring operation is a backup part of that valve for the direct acting-type valve.

MEMBER ABDEL-KHALIK: Okay.

CHAIR CORRADINI: But that's not a set, as I remember in somewhere in your description, that's not a setpoint in some sort of succession, correct?

MR. DEAVER: Yes. Yes, that's true.

CHAIR CORRADINI: Okay.

MR. DEAVER: I just wanted to point out that those would actuate first.

Okay. This shows the arrangement of SRVs and SVs and the DPVs. Basically the longer steamlines of accommodation of five valves versus the shorter steamline has four valves. And so we have a mixture of SVs and SRVs.

I think the key point was to distribute them fairly equally so that if there was an issue with any given steamline, that you would get both types of valves in operation.

The PBVs are shown as separate. They're not on the main steamline at this point.

Do you have anything to add to that, Joel or --

MEMBER ABDEL-KHALIK: I was just wondering about -- I mean in the diagram that we have, you know you have an asterisk indicating which one are designated as ADDs. And I was just trying to figure out the logic of why these particular ones are designated as such. Because they're not symmetric.

MR. DEAVER: Yes.

MEMBER ABDEL-KHALIK: The longer lines have three, the shorter lines have two and they're not exactly the same two. Does that produce sort of asymmetric loading?

MR. DEAVER: As far as if you were to actuate them?

MEMBER ABDEL-KHALIK: Right.

MR. DEAVER: That shouldn't make any real difference, you know, on the actuation part. This would probably have more impact on things like the acoustic loads and stuff on --

MEMBER ABDEL-KHALIK: Well, that's what I meant.

MR. DEAVER: But as part of the dryer program, they are basically arranging -- you know, the valve and the sand pipes are all the same. So the signals and such that come from the SRVs are fundamentally the same.

So the mix of SRVs and SVs are not really significant. But the location to detune them such that they don't send reenforcing signals to the dryer is important. And that's part of the dryer program.

MEMBER ARMIJO: Okay. That may address a question I have. You mentioned in the DCD that you're arranging the SRVs and the DPVs to minimize something called simmering. I don't know what simmering is.

MR. DEAVER: Okay.

MEMBER ARMIJO: So I don't know whether it's a good thing to minimize it or a bad thing. Could you explain that?

MR. DEAVER: Joel, you want to talk about simmering?

MR. MELITO: Simmering is essentially referring to the fact, and this is somewhat of a problem with the older BWRs, is the relationship of the actual mechanical setpoint, the pressure lift of the safety valve relative to the normal operating pressure of the plant. And the closer that setpoint is to normal operating pressure, the less stable the valve is. So it has a tendency in some valves to kind of just sit there and chatter on its seat and leak steam into the containment in that way. So we've tried to in this design push those setpoints another 100 psi higher to get more simmer margin and eliminate that problem to the best that we can.

MR. DEAVER: Yes. We prefer not to have any simmering as a design objective.

MR. MELITO: Yes. Now the DPVs themselves will not simmer. They are essentially a hermetically sealed valve. They do not have a simmer margin.

MEMBER ARMIJO: Okay. Okay.

MEMBER ABDEL-KHALIK: But wouldn't these valves sometimes float on their setpoint and go open and closed?

MR. MELITO: Well, keep in mind that for the design of the ESBWR most of the pressure response in transients is going to be carried by the ICS, the isolation condenser system and is not expected, in fact it's purposely designed that the SRVs and the SVs do not lift. In fact, the peak pressure does not approach close enough to begin to cause them to lift. We try to maintain enough margin to avoid any anticipated reduction in setpoint that might inadvertently occur away from the nominal setpoint we've allowed for that to prevent that from happening.

MEMBER MAYNARD: Is there a reason that the DPVs are separate penetrations and weren't put ont he steamlines there? I'm just curious.

MR. DEAVER: Well, actually because of steam dryer issues and we felt it was better to not have them associated with the steamline. It just takes away another element that could cause signals or acoustic loads on the dryer. So we felt it was better to divorce it from the steamline.

MEMBER MAYNARD: I'd just take it at penetration, but that's --

MR. DEAVER: Well, we already have core penetrations for the iso-condensers anyway.

MEMBER MAYNARD: Okay.

MR. DEAVER: So --

CHAIR CORRADINI: Oh, so these are taken off of an elbow that's going to go to the isolation condensers anyway?

MR. DEAVER: Yes. Right.

MEMBER MAYNARD: Okay.

CHAIR CORRADINI: I got it. I got it.

MEMBER ABDEL-KHALIK: Is there empirical evidence that supports your selection of these penetrations with regard to the impact upon the steam dryer, or this is just gut feeling?

MR. DEAVER: Well, initially in our initial design we had some DPVs on the steam line, and they were right at the initial horizontal line coming out of the vessel. It would have placed them very close to the steam dryers themselves. And so we didn't have any real evidence that that was going to be a problem, but it's just another unknown that we didn't want to introduce into the system.

MEMBER ABDEL-KHALIK: Well, my question I guess, do you have any modeling capabilities that would allow you to predict that a priori?

MR. DEAVER: Well, we've done modeling, what we call scale model testing on the dryer program. We will be doing some scale model testing. But at this point our plan is not to have the DPVs in the steamline itself.

MEMBER SHACK: To presumably do the scale model testing for the SCs --

MR. TUCKER: Jerry?

MR. DEAVER: Yes.

MR. TUCKER: This is Larry Tucker.

Could you go back to your simplified drawing where you show the DPV on the ICS system?

MR. DEAVER: Okay.

MR. TUCKER: Note that the DPV -- going farther back, your original one.

MR. DEAVER: Yes.

MR. TUCKER: Is on a system that

essentially has no flow in it.

MR. DEAVER: Yes. We call this the specs.

MR. TUCKER: And therefore the acoustic loads, it doesn't generate acoustic loads since there's no flow.

MR. DEAVER: The operation system there's no flow in the normal operation.

MR. TUCKER: And that's a large part of -- the flow is the driving force that creates the load. So if you can remove the flow going past the valve, then you can remove the potential load. And so that plays into the rationale for why it's placed there.

The placing of the SVs and SRVs on the main steamline, now we're back into familiar territory for the rest of the BWR fleet.

And your question of methods. Yes, there's CFD analysis and other tools that we use. So I won't go into all of them here, but they're common tools.

MR. DEAVER: Okay. Moving along then, next we talked about materials in the reactor coolant pressure boundary. And basically all the materials that we're using are familiar materials that we've used in the past. The main difference is in the feedwater line. We were planning to use a low alloy material, a P22 material, to provide more corrosion resistance and to counteract any FAC issues. So that system even going beyond the containment is a low alloy material also.

So, you know piping components, fittings are all typical.

The iso-condenser tubing, we use a modified Alloy-600 material for the tubing. And by "modified," we mean a niobium-stabilized material which adds corrosion resistance. This is a method that was developed in Japan and demonstrated to be corrosion resistent. And so that anywhere we use inconel materials we plan to use the niobium-stabilized materials.

MEMBER SHACK: What's that used in the Japanese ABWR?

MR. DEAVER: Well, any inconel application is a step tube --

MEMBER SHACK: Is step tubes? Yes, okay.

MR. DEAVER: And in the support for the strut support; those are typical uses. And then strong head bolts, the main stud or the shaft on the bolts. That's typical uses of inconel.

MEMBER ARMIJO: So this niobium modified inconel has been used in Japan in the ABWRs?

MR. DEAVER: Yes.

MEMBER ARMIJO: Or how about in the U.S., any U.S. experience?

MR. DEAVER: Have we used in any, Brian, that you're aware of?

MR. FREW: No. Actually, other than the shroud head bolts.

MR. DEAVER: Yes. Yes, shroud head belts.

MR. FREW: That have been installed.

MR. DEAVER: Right.

MR. FREW: I mean, it's a material that's used, I mean in the construction of the new reactors as far as it has been applied.

MR. DEAVER: Yes. We did apply it to the one reactor.

MR. FREW: Okay. So it's --

MR. DEAVER: It's still in construction.

MEMBER ARMIJO: But actual service is in the first ABWRs? Did the first ABWRs have these materials?

MR. DEAVER: Yes. Yes, definitely had that. Yes.

MEMBER ARMIJO: At about what, ten years or more?

MR. DEAVER: Yes. The initial ABWR, the K6 reactor has that, that's ten years.

MEMBER ARMIJO: Anybody will make it if you pay for it.

MEMBER SHACK: Just one of the things that always -- you know, I didn't see any sulfur specs on any of your carbon or low alloy steels. And, you know, one thing we sort of learned is that, you know, sulfur is not a particularly good thing to have in these systems. The Japanese always have very slow sulfur steels for those applications. I assume that somewhere you really intend to keep the sulfur levels down. I couldn't find a word about sulfur anywhere in the material specs.

MR. FREW: Yes. I mean for the primary carbon steel materials we do control the sulfur. And--

MEMBER SHACK: But I mean, you know the spec that you've stated there certainly will let you have all the sulfur in the world.

MR. FREW: The plan is to ,010 is the limit.

MEMBER ARMIJO: For the carbon steel?

MR. FREW: For carbon steel, yes.

MR. DEAVER: It's in our generic, you know project material spec. But it hasn't been introduced into the certification document.

MEMBER ARMIJO: So it will be less then or equal to 0 point --

MR. DEAVER: What was it again, Brian?

MR. FREW: .010.

MEMBER ARMIJO: 010.

MEMBER SHACK: Beaucoup sulfur by Japanese standards.

MR. DEAVER: That's a lot of sulfur by Japanese standards.

MEMBER ARMIJO: Yes, as long as we're talking about sulfur, there's been some work, I think probably industry work done that NRC Research reported on it that very low sulfur in addition to the low carbon was beneficial as far as IGSCC. And I was wondering if GE-H was going to specify very low sulfur stainless steels for their core internals or other components as part of the ESBWR?

MR. FREW: Yes. I mean, it will be as specified in our project documents. So I can't tell you an exact number at this time.

MEMBER ARMIJO: Okay. If you could just get it to us later, that would be fine.

MR. FREW: Okay.

MEMBER SHACK: Yes, what's the difference between a project document and an ICD?

MR. FREW: Well --

MEMBER SHACK: You don't want to commit to some of these things but you're really going to do it?

CHAIR CORRADINI: I think you're putting words in their mouth.

MR. DEAVER: I guess if we needed to talk them into it, we could. I mean, it just been brought up.

MEMBER MAYNARD: Well, that's really not unusual, though to have higher --

MR. TUCKER: This is Larry Tucker.

MEMBER MAYNARD: -- it's just licensing.

MR. TUCKER: What they're referring to are project materials specifications that are at a different level of detail than the design certification document. We have valve specification, pipe specification, electrical cable specification. So it's a question of level of detail. It's not that we don't have it, it's just that it doesn't rise to the level of detail to be included in the DCD.

MEMBER ARMIJO: Well at some point I, for one, would like to see the specs that would be used for the materials for this plant.

MR. TUCKER: We'd be happy to do that.

MEMBER ARMIJO: Yes. I think if we had them, we probably wouldn't be wasting so much time on these.

I had a question on the carbon steels in view of the flow-accelerated corrosion event, I guess Japan. Is the steamline or all the other carbon steel lines --

MEMBER SHACK: The steam water is now a P22.

MEMBER ARMIJO: Yes. That's --

MEMBER SHACK: That's two and a quarter chromium molly.

MEMBER ARMIJO: Well, they have a -- okay. So that's two and a quarter?

MEMBER SHACK: Yes.

MEMBER ARMIJO: Is that correct?

MR. DEAVER: Yes, that's correct.

MEMBER ARMIJO: Two and a quarter? What about other steam?

MR. DEAVER: Well, the steamline in the RWC lines, which are carbon steel, the reason we went to the low alloy on feedwater was because of the flow rate in that line.

MEMBER ARMIJO: Okay.

MEMBER SHACK: And it's water.

MEMBER ARMIJO: It's water, right.

MEMBER SHACK: It's water.

MEMBER ARMIJO: It makes a difference.

MR. DEAVER: Yes.

MEMBER ARMIJO: Okay. So you have addressed both?

MR. DEAVER: Well, we have internally evaluated that on steamline and so forth. Determined that we didn't need to upgrade in that line.

MEMBER ARMIJO: Yes. There was another thing here that I was puzzled in reading the document. Is that there seems to be a disagreement between the Staff and GE-H on calculating the amount of delta ferrite for the cast stainless steels. And it bothers me that this is even an issue, that it's such a small -- I don't understand why the GE-H wouldn't simply use the Staff's methodology.

MR. DEAVER: Well, we are at this point.

MEMBER ARMIJO: Oh.

MR. DEAVER: We've committed to doing that. We just need to respond to a -- at this point.

MEMBER ARMIJO: Okay.

MR. DEAVER: We consider that a resolved item.

MEMBER ARMIJO: Oh, okay. Well, then I'm

not going to raise it again.

MR. DEAVER: Okay. Okay. The next slide just lists materials for valves and also for the pressure vessel. These, again, are just typical materials that we used in prior plants.

One difference, and I'll point it out when I get to the vessel prong, is that we're using larger ring forging for the first time in the U.S. It's been used in Japan and so forth. But that's a significant upgrade that we made to the vessel design.

MEMBER SHACK: Now is this vessel going to come in two pieces to the site?

MR. DEAVER: No. No.

MEMBER SHACK: It's going to come --

MR. DEAVER: One piece.

MEMBER SHACK: One piece.

MR. DEAVER: Well, even in a worst case scenario, which was the North Anna site, which is inland about 85 miles, we've done a study and found that it's feasible to bring it in one piece. So --

CHAIR CORRADINI: And one more time, how big is this thing?

MEMBER ARMIJO: Nine feet tall.

MR. DEAVER: It's a 1000 ton, and it's --

CHAIR CORRADINI: The length is what I'm--

MEMBER SHACK: A big truck.

MEMBER ARMIJO: It's 91 feet or something like that.

MR. DEAVER: Yes.

MEMBER ARMIJO: Thirty meters.

MR. DEAVER: Yes, 90 feet or so, roughly. It's --

CHAIR CORRADINI: This is without it's top, right?

MR. DEAVER: Without the head on it.

The increase in length is 6½ meters because of the natural function in the reactor.

MEMBER ARMIJO: Diameter is the same as the ABWR?

MR. DEAVER: The diameter is the same as ABWR.

MEMBER ARMIJO: So the ring forging technology is not new?

MR. DEAVER: Essentially they are the same, basically.

MEMBER ARMIJO: A few more down in the core region.

MR. DEAVER: Yes.

MEMBER ARMIJO: Yes. Big sucker.

MR. DEAVER: Yes.

MEMBER ARMIJO: Well, that takes care of my questions on the -- I was going to ask you how are you going to build that on site, but I'm not going to ask it now.

MR. DEAVER: Right. Okay.

I've included a slide that talks about stress corrosion in stainless steel materials. Basically they revolve around avoidance of sensitization, which we control by carbon content and other process controls. We also do sensitization testing and IGAs testing, which is standard to validate that the materials are not sensitized or potentially can be sensitized.

And then the second bullet basically focuses around contaminates during fabrication and so forth, the effective cleaning and preventing of in-process materials coming in contact that have high sulfur and phosphorus and the known contaminants.

And then the last item deals with cold work. So we control that by hardness tests and so forth. And we basically limit and control any grinding processes. Vendors have to do a qualification program on any grinding process. And then we control it by surface finish and so forth.

MEMBER ARMIJO: You don't prohibit

grinding, post-weld grinding?

MR. DEAVER: It is almost virtually impossible to prevent it. But we try to minimize it.

MEMBER ARMIJO: My understanding that the Japanese managed to crack 3.15 nuclear grade shrouds by post-weld grinding.

MR. DEAVER: Yes. That was grinding that wasn't controlled in any manner.

MEMBER ARMIJO: I don't think there's anyway you can control it and make it acceptable. But that's an economic risk.

MR. DEAVER: Yes.

MEMBER ARMIJO: And I'm surprised that GE doesn't just simply prohibit it. In order to get a good x-ray you create a stress corrosion problem downstream. There's got to be a better way. But you don't prohibit it?

MR. DEAVER: No, we don't at this point in the bed.

We do things like try to minimize number of welds and NTUs --

MEMBER ARMIJO: Yes. Yes. Okay. I'm just not going to tell you how to do your job.

MR. DEAVER: So this summarizes some of the major aspects that we control for stainless steel.

Okay.

MEMBER ARMIJO: Now you use this sensitization test, it's just an acid test, right, that ASTM special sensitization test you mentioned?

MR. DEAVER: Yes.

MEMBER ARMIJO: You know, that's pretty much outdated? It's an antiquated test that for material to fail that test, it has to be grossly maltreated. I wonder why you just don't use something more modern?

MR. FREW: I guess my question is which? Are you referring to the practice E-sulfur acid test?

MEMBER ARMIJO: Yes. Yes.

MR. FREW: The test we actually use is the modified practice A with the 5 percent ditching limit. So we have a tighter limit. It's mainly to show that the material was treated properly.

MEMBER ARMIJO: But you also can actually do an IGSCC test in high temperature water. You have much more direct testing to show that the material is good as opposed to these acid tests which have never really been particularly useful. And I just wondered why you use the ASTM test instead of something, you know, that actually can produce IGSCC?

MR. FREW: Well, I think that what we're

trying to show is was the heat treatment done properly. And it really -- I mean the test you're talking about is a constant extension rate test, which it would be a rather extensive testing program to do that.

I mean, it's our belief that material that has been subjected to this modified practice A, not the standard one where you can have fully grain boundaries surrounded, does show that the material is acceptable.

MEMBER ARMIJO: Okay. That may be valid if you've got a data to show that acceptable -- if it's acceptable by virtue of this modified acid test. It still uses acid, right?

MR. FREW: Yes. Yes. Ten percent--

MEMBER ARMIJO: Yes. But you've done, let's say, constant extension rate testing in oxygenated high temperature water and show that, in fact, it's a perfect predictor of IGSCC resistance, and I'd be happy. But I don't know if you have that data and that the Staff has received that data.

MR. FREW: I'd have to go back and review it --

MEMBER ARMIJO: Okay.

MR. FREW: -- to locate that type of

information.

MEMBER ARMIJO: My guess is it doesn't exist.

MEMBER SHACK: But, I mean they're really using as a QA test. I mean, their real reliance is on the low carbon level and the fact that they're not going to sensitize it.

MR. DEAVER: Right.

MEMBER SHACK: And that would mostly tell you if somebody happened to ship you the wrong material. But, I mean, you could do an EPR test if you want a quick test that most people would think is a better test to sensitization.

MEMBER ARMIJO: Yes, I think I guess that's really my bottom line. I just think some of these antiquated ASTM tests should be put out to pasture now that we know so much more IGSCC.

MR. DEAVER: Okay. Moving to the next topic if that one's finished. The next one is on the link detection and isolation system. Basically, these are the automatic isolations that are designed for ESBWR. These are all standard except for the ICS actuations.

In the second bullet we have the case where steamer condensate flow is occurring. That would be an indication that the system's open, so that would cause an automatic isolation.

And then ICS radiation, that would be radiation coming from the pool area where the condenser is itself. If we detected radiation in that area, that might either be an indication that there's a tube leak or that there's a flange leak in the system. You know, just a mechanical leak through a gasket.

So each of those would cause an isolation.

MEMBER ABDEL-KHALIK: So ICS actuation can actually happen before MSIV closure?

MR. DEAVER: Well, that would have to be a spurious type of thing that would happen. It wouldn't be a design situation. We wouldn't normal actuate ICS unless there was a containment isolation. An inverted --

MEMBER ABDEL-KHALIK: So when you say automatic isolations on ICS steam condensate flow, what does that mean?

MR. DEAVER: That means that the system's been open.

MR. MELITO: That's break flow.

MR. DEAVER: Pardon me?

MR. MELITO: That's break flow.

MR. DEAVER: Oh, okay. I'm sorry. Let me clarify that.

This is like the other cases like main steam and reactor water cleanup where there's actually a break in the line. And this would be an indication that there's a break. And for that reason we need to isolate the containment.

I misinterpreted that requirement. Okay.

MEMBER ABDEL-KHALIK: And before you move onto section 5.3 could you tell us about your unidentified leak break limits?

MR. MELITO: Yes. What we have in the current specification or the certification is the 5 gpm, which is the traditional leakage that we've had. Actually, it was developed at the time that we started getting leaks or cracking in the recirc lines. And that was developed as a credible leak rate, you know, for unidentified leakage in that time frame, and that's been carried forward in all the plants. And--

MEMBER SHACK: I thought lots of plants actually went to 3 gpm when they were running -- you know, they hadn't fully modified their cracking.

MR. DEAVER: Well, that may have been a temporary condition before they did an implementation of --

MEMBER SHACK: I see.

MR. DEAVER: -- better processes.

So this is what we call the tech spec limit where we would actually have to initiate actions at 5 gpm.

MEMBER ABDEL-KHALIK: Now the document I have here, which is Rev. 3, I guess, says something about some instrumentation activating an alarm in the main control room at 25 gpm.

MR. DEAVER: Well, that's the identified leakage rate. You know, there's the two types.

MEMBER ABDEL-KHALIK: Right.

MR. DEAVER: One is identified versus unidentified. So that 25 is associated with the identified leakage.

MEMBER ABDEL-KHALIK: So how is the level of unidentified leak determined?

MR. DEAVER: Well, it's a combination of collection of water in sumps. And there's a reg guide that gives a lot of criteria on radiation release and moisture separation detection in the system.

MEMBER ABDEL-KHALIK: So this is not something that's automatically sort of indicated to the operator?

MR. DEAVER: Well, what we've committed to in the last revision is a process for determine actual leakage, you know, using these different inputs. So that's --

MEMBER SHACK: I mean, leakage is unidentified until you identify it?

MR. DEAVER: Yes. Right. Well, from the unidentified sources we will determine a leak rate which will be known to the operators. And then they'll start to take measures to determine what the cause is at that point.

I should clarify that plants always have some amount of unidentified leakage, and that's been our main concern in changing that leak rate is that we just have a baseline amount of leakage within the system that can't be avoided. And so we hate to put the detection level too close because then it just becomes more of a nuisance item when we know it's just standard kind of situation during operation.

MEMBER ABDEL-KHALIK: Now in this document you often refer to sensitivity versus accuracy of these unidentified leaks as being 1 gpm, and you use them interchangeably. What do you really mean? Is it sensitivity or accuracy?

MR. DEAVER: Well, part of it is the quantifying of the amount, and then accuracy has got to do with the calibration of the system.

MEMBER ABDEL-KHALIK: So both of these are 1 gpm?

MR. DEAVER: Well, the accuracy of the system is not -- well, I think combined we have a detection level of 1 gmp. Yes. Within one hour. So that's the limits that we can start accurately identifying leakage is at 1 gpm.

MEMBER ABDEL-KHALIK: So when you talk about sensitivity, what does that mean?

MR. DEAVER: Does that have a special meaning to you, Joel?

MEMBER MAYNARD: Well, because of the nature of the containment in part the sensitivity is good enough to be able to accurately measure a change in leak rate of 1 gpm in one hour. And this has been the issue. But this works best if you had a dry containment.

The problem we face with the containment for BWR is there are always large amounts of water available that can upset and give you a kind of a background noise that kind of masks what's really leakage and what's just due to temperature change or some other evolution that's going on.

We have this large suppression pool and we actually in this design have additional open pools of water that can evaporate and come back into the pools. Effectively all will find their way into the sump inadvertently and it gives you a false signal that there's a leak rate, and it's just water moving around inside the containment itself.

So you have a problem in how sensitive a high precision instrument can be. It can be 1 gpm, but that's in a dry environment. In a BWR environment we're going to have to deal with the fact that there's some other things going on that is not real leakage from the RCS anywhere, or anywhere from the reactor coolant pressure boundary. And that's a discussion we've been having with the Staff. We still have to address a question on that.

MEMBER ABDEL-KHALIK: So is that the reason why the unidentified leak rate limit is set at 5 gpm?

MR. MELITO: Yes, it is.

MEMBER ABDEL-KHALIK: Thank you.

MR. DEAVER: Okay. The next section is -- let's see. I went back. The next section is 5.2, which deals strictly with the reactor pressure vessels. And there's three main topics covered.

One is vessel materials and processes.

The second deals with the pressure temperature limits that deals with the fracture mechanics aspects of low alloy materials. Basically we follow all the guidelines of 10 CFR 50 Appendix G and the Reg. Guide 1.99. So we have full compliance there.

And then as far as reactor vessel integrity, we use proven materials and fabrication methods which have been proven over time, you know consistent with the ASME code. NDE inspections are consistent with the code and have been reliable. And then we have a surveillance program where we have materials that are sampled in the operation of the plant that give us an indication of embrittlement of materials.

MEMBER ABDEL-KHALIK: If I may go back to the leak rate issue, there's a statement here that says that the unidentified leakage rate limit is based with an adequate margin for contingencies on a crack size large enough for leakage to propagate rapidly. How details is that analysis and does it account for all anticipated loading conditions that you might expect in various sites where you can have a crack?

MR. DEAVER: You want to answer that, Joel?

We did just a representative analysis to look at that.

MEMBER MAYNARD: Yes. The size of crack that you need is actually quite small, 4 or 5 gpm. And actually, a more typical number for the first components that we would be concerned about that are approaching a crack condition that we would consider critical, it probably would be leaking at a rate of 25 gpm or more. So we have this lower limit which gives us lots of margin between where crack sizes, even for some piping, would become a concern that there might be a continuation of that crack around to a separation of that pipe versus what we can actually detect with a lot of confidence.

MR. DEAVER: Yes. To some degree because we're not like the PWRs or the AP1000 where they're depending on leak before break, you know we've been somewhat resisting doing a full blown fracture mechanics analysis to determine critical flaw size because we design for pipe breaks. But we have done some scoping work to determine how that relates, the 5 gpm relates to an actual leak rate. You know, the integrity of a pipe, the major piping.

MEMBER ABDEL-KHALIK: So does this statement refer to a detailed fracture mechanics type analysis or is this sort of more of well how big of a crack do I need to get 5 gpm?

MR. DEAVER: Yes, it's the latter.

MEMBER ABDEL-KHALIK: Okay. Thank you.

MR. DEAVER: Yes.

MEMBER SHACK: Just on this inspection for the reactor vessel, how much room do you actually have now to perform these inspections on this vessel?

MR. DEAVER: Pressure vessel welds?

MEMBER SHACK: Yes.

MR. DEAVER: Well, older reactors we actually hung the insulation off the vessel. But the more typical ones and BW-5s and 6s the insulation is hung off the shield wall. So we typically have 18 inches or two feet of space. We typically install remote crawling devices that contract the welds.

It's not a problem on later BWR designs, including ESBWR.

Okay. I wanted to go through the vessel, some of the key things that will be changed.

The thing I would like to first point out is I talked earlier about large ring forgings. What we've done is we used large ring forgings for the vessel flange on the upper end -- on the head flange and vessel side -- and then we have forgings that start in the area where the vessel support is located. We have a shell ring here, a forging ring, and then we have one long forging that covers the core belt line region. And then we have a forging for the transition piece to the bottom head dome.

So all the penetrations in the vessel bottom head are through this bottom head disk forging and so there's no weld seams associated with the pattern of CRDs and so forth.

MEMBER ARMIJO: Where are the welds in relationship to the core? On the ring -- yes, that one.

MR. DEAVER: Yes. The main ring forging is slightly above top of active fuel and slightly below the bottom of active fuel location. It's four meters in length --

MEMBER ARMIJO: Okay. Which is --

MR. DEAVER: -- for that core belt line forging. And that's the maximum length --

MEMBER ARMIJO: A little bit longer than the fuel?

MR. DEAVER: -- we can produce. Okay?

As mentioned before, the vessel is 6½ meters taller. And that's primarily because of the need for the natural circulation function. We have a chimney section which facilitates the pressure differential to cause natural circulation. And then we have chimney partitions which direct flow up into the separator and dryer itself.

Basically the components like the steam dryer, the separators, top guide core plate shroud are very typical of what we've had in the BWRs as well as the CRD and in-core penetrations. So to the maximum extent possible we have kept the component designs as typical from a performance viewpoint.

Obviously for the dryer we're going to make it more robust to withstand all the different kind of loadings will be post on it.

Do you have a question or --

MEMBER ABDEL-KHALIK: How many ring forgings now do you have on this?

MR. DEAVER: Well, we have six forgings that are large ring forgings. We have four associated with this bottom head region. So we have one in this region. Two, three and four for the dome. And then we have two in the upper area.

In the middle we use plate materials and we use two halves to form a shell. And we have five sections that are in the shell sections.

And then the dome is made from fabricated plate also that's formed.

MEMBER ABDEL-KHALIK: Okay. Now somewhere in here it says that there are capsules provides to consider the 60 year design life of the vessel.

MR. DEAVER: Yes. Yes. We have surveillance capsules that are positioned opposite the core region.

MEMBER ABDEL-KHALIK: What if somebody wants to extend this life beyond 60 years?

MR. DEAVER: I'll Brian address that. We would need more specimens.

MR. FREW: I mean, in the practice with today's reactors is to include reconstituted capsules in the vessel. They've back and recalculated the Charpies and placed them in again to account for the extra time.

MEMBER ARMIJO: Yes. But you can't make up for time, though. I mean if you --

MEMBER SHACK: If he takes it out, he busts it, then he puts it back together and puts it back n.

MEMBER ARMIJO: He takes an irradiated material and puts it back in? Well, that'll work.

MR. FREW: They'll be out of the vessel for --

MEMBER ARMIJO: It'll be smaller.

MR. FREW: -- one cycle, and then placed back.

MEMBER ARMIJO: Okay. That's another way to do it.

MEMBER ABDEL-KHALIK: But if you want to hit a time that this is a possibility for people to go to 80 years, why not take that into account from the very beginning?

MEMBER ARMIJO: Another capsule.

MR. DEAVER: I think we're getting to the point of where we've got extensive number of specimens.

MEMBER ARMIJO: More specimens than vessel.

MR. DEAVER: It's more better, I guess, at this point.

MEMBER ARMIJO: It might be prudent.

MR. DEAVER: I guess that's coming up on operating plants as an issue of possible.

Okay. Well that covers most of the key areas. We've made some minor modifications to the vent system. It's been a problem in service to disconnect piping, so we have a vent system that comes down from the vessel head to the main --

MEMBER ABDEL-KHALIK: I have a question as to --

MR. DEAVER: Yes.

MEMBER ABDEL-KHALIK: -- the mechanical coupling. Is the chimney rigidly coupled to the dryer?

MR. DEAVER: The chimney is down here.

MEMBER ABDEL-KHALIK: Right.

MR. DEAVER: We have a barrel section --

MEMBER ABDEL-KHALIK: So it's hanging from someplace or it's bolted?

MR. DEAVER: It sits -- it actually connects onto the top of the top guide.

MEMBER ABDEL-KHALIK: Okay.

MR. DEAVER: It sits on there to take a continuous flow through the core up through the channels.

MEMBER ABDEL-KHALIK: But there is no direct mechanical coupling between the chimney and the dryer?

MR. DEAVER: No. The dryer's up here. It has a skirt. It sits on its own support rack. And so it's independent of the separators or any structure below it.

MEMBER ABDEL-KHALIK: Okay.

MR. DEAVER: Okay. And one thing that

we've done instead of four brackets typically on the steam dryer support, we've gone to six now to give it better support.

MEMBER ARMIJO: Are you doing anything different about -- with the stainless steel cladding of the vessel than you've done before, on let's say ABWR vessels?

MR. DEAVER: That process is basically the same. They use strip cladding.

MEMBER ARMIJO: Okay. So there's nothing different that would be specified for this vessel?

MR. DEAVER: No. Not that I'm aware of.

MEMBER ARMIJO: And is it entirely clad or is it just partly?

MR. DEAVER: All the interior surfaces of the vessel are clad except in regions of some of the nozzles. We typically don't worry about the nozzles. We prefer to be able to have good access for UT inspection and so forth.

We typically have stainless cladding all the way down the cylindrical section. But when we get into the bottom head where we have inconel stub tubes, then we use inconel cladding.

And typically the head is not clad because it's in a steam environment.

MEMBER ARMIJO: Yes.

MR. DEAVER: So that's --

MEMBER ARMIJO: So you've used the inconel cladding on the bottom heads before?

MR. DEAVER: Yes. That's been standard ever since we've been using the inconel stub tubes.

MEMBER ARMIJO: Okay.

MR. DEAVER: Okay. I think those are the major points.

The next slide is kind of just a summary of things I've talked about.

Basically the core region is two foot shorter because of the shorter fuel.

And we talked about the chimney. The delta that changes in height. The large ring forgings. The vent system is fundamentally the same, although it's routed differently.

The one thing I didn't mention is for ABWR we have the steam flow line restrictor is actually part of the vessel design. It's in the nozzle itself.

CHAIR CORRADINI: Right at the boundary?

MR. DEAVER: Well, right here. See, the forging that is welded into the vessel shell, the restriction is right in the throat of this venturi here. So that if there was a steamline break, this now becomes the limiting flow feature.

MEMBER ABDEL-KHALIK: And how big is the throat?

MR. DEAVER: It's about 14 inches.

So this also serves as a venturi here for measuring steam flow. So we have a tap that comes in near steam flow from that. That was a feature that was adopted for ABWR, but not prior reactors.

Okay. On materials and process controls on RPB, basically we follow the ASME code on design and material requirements. Because of the concern of some elements, materials we limit to copper, both phosphorous and nickel content which is pretty standard in the industry to minimize fracture toughness.

MEMBER ARMIJO: Maximize.

MR. DEAVER: Well, yes, minimize.

MEMBER ARMIJO: Minimize embrittlement.

MR. DEAVER: Right. Post-weld heat treatments are standard. All components or all welds are post-weld heat treated on the vessel.

And we specify Rt_{ndt} properties of minus 20 degrees, or lower. Actually with the large ring forgings we typically do better than even that standard.

MEMBER ABDEL-KHALIK: Now what is the expect Rt_{ndt} after 60 years of operation?

MR. DEAVER: Is it 60?

MR. FREW: The right hand of the calculated shift, I believe -- actually, Jerry, it's on the next chart.

MR. DEAVER: Is it? Okay. Okay.

MR. FREW: Thirty-seven degree celsius.

MR. DEAVER: Yes. I included this slide. This shows the perfect temperature curve for hydro test purposes. It establishes what the vessel temperature has to be for a corresponding pressure in the vessel.

This is a typical curve that's generated per the standards.

The only point I wanted to make here is that this is a representative curve at this point. What we do for every vessel is establish a separate curve based on the actual vessel material properties, which will be done after we've been able to fabricate the vessel. Be provided with the vessel.

Section 5.4 includes a lot of topics, some of them are relevant to ESBWR. And we've covered several of these already.

Basically the reactor recirc system is

really the natural circ system and as such, there are really no components that form that system. It's just basically a product of the vessel design and so forth.

Reactor coolant piping. We've already established that we don't have any major piping below core, which is a safe feature associated with the ESBWR.

We've talked about the flow restrictors being a part of the main steam nozzle.

The main steam isolation valves are basically the system that provide isolation for the steam lines. Those are -- well, typically what we describe in the control document is the wide globe valves at this point.

We've talked about iso-condenser system.

CHAIR CORRADINI: Can I stop you there?

MR. DEAVER: Yes.

CHAIR CORRADINI: We were kibitzing over here.

So the bold and no bold is, the bold is the RCS, the nonbold is stuff hooked onto the RCS?

MR. DEAVER: Well, I basically bolded items that I wanted to discuss in a little more detail--

CHAIR CORRADINI: Okay.

MR. DEAVER: -- versus stuff that I thought I'd already covered.

CHAIR CORRADINI: That's fine. So let's just stay the isolation condenser for a moment. So I thought it was part of the RCS, yes?

MR. DEAVER: Oh, it is, yes.

CHAIR CORRADINI: Okay.

MR. DEAVER: Most definitely.

CHAIR CORRADINI: All right. Okay.

MR. DEAVER: No, I wasn't trying to distinguish what's in and out.

CHAIR CORRADINI: No, no. I just was trying to understand the --

MR. DEAVER: Right. Just a clue for me as to what to talk to.

CHAIR CORRADINI: Fine. Thank you. Sorry.

MEMBER ARMIJO: Well, are you going to talk some more about the isolation condenser system?

CHAIR CORRADINI: I don't think so.

MR. DEAVER: I hadn't planned to, but if you've got more questions.

MEMBER ARMIJO: Yes, I have. And that's the materials in the isolation condenser.

MR. DEAVER: Okay.

MEMBER ARMIJO: And that's the extent to

which you have experience with these materials; the water chemistry on the primary side and the secondary side? You know, can you just elaborate on that a little bit?

MR. DEAVER: Yes. Inside the pipe will be steam and water combination.

MEMBER ARMIJO: Right.

MR. DEAVER: And outside we have a pool with demineralized water so it will be high purity water also.

MEMBER ARMIJO: Yes.

CHAIR CORRADINI: And you're going with inconel?

MR. DEAVER: Yes.

CHAIR CORRADINI: You said it, I know it's somewhere in here you said?

MR. DEAVER: It's the inconel Alloy 600, the modified.

CHAIR CORRADINI: Okay.

MEMBER ARMIJO: And that's based on -- and you have the Japanese experience?

MR. DEAVER: Yes.

MEMBER ARMIJO: And isolation condenser application or not?

MR. DEAVER: They tested reactor

conditions, which --

MR. FREW: The use of modified 600 for the Japanese has been for the vessel. In the case of the isolation condenser it has not been -- this particular alloy has not been used because this is the first application of an isolation condenser since the early years. But I think to answer that, one of the plants, Millstone I believe, replaced their isolation condenser because of problems they had with the stainless. And they actually used the ordinary Alloy 600.

And my understanding was there weren't any further issues based on the --

CHAIR CORRADINI: But that's a change done one of the current BWRs? Current?

MR. FREW: Yes, in the past. I mean, that was done a long time ago.

MR. DEAVER: Yes. It's a plant that's not in operation.

MEMBER SHACK: But what does Nine Mile and Oyster Creek for their stainless steel, I assume?

MR. FREW: They have.

CHAIR CORRADINI: Are you done? I didn't mean to interrupt you.

MEMBER ARMIJO: Yes. They answered my

question.

CHAIR CORRADINI: I want to ask this question. So we'll be able to discuss the isolation condenser in other settings, I assume, or is this our last portion?

MR. DEAVER: It comes up-- well, this may be the main opportunity.

CHAIR CORRADINI: Okay. So then I'll ask my question now.

MR. DEAVER: Okay.

CHAIR CORRADINI: So what are the lessons learned from the old isolation condensers to this design? What's the delta in the design here that you would consider, or are they similar? Because I'm still back to my delta into what I've got operating versus what I have here. And you've mentioned materials. I wanted to know if there was something else that I may have missed.

MR. DEAVER: Well, the one thing I wanted to mention is that in this design we don't have any crevices associated with the welding of tubes to pipes and so forth.

Our main experience with inconel in the past has been where we've had crevice conditions or filler metals that have been with flux.

MR. TUCKER: Jerry?

MR. DEAVER: Yes?

MR. TUCKER: This is Larry Tucker.

Hugh Upton was involved in that. Maybe Hugh can talk about what we've done in the test program.

MR. UPTON: Yes. Just let me bring everybody up to speed on the design of the IC.

The IC for ESBWR, although the concept was used in the early BWRs, the configuration of the IC is different. We have done full scale testing in Italy at Fiat of the IC system that's been designed. So we have assembly drawings already. And we did test it with the inconel. So we've got very successful test data based on those testings.

And we've changed the material from stainless steel to inconel.

CHAIR CORRADINI: So it is a different design?

MR. UPTON: Yes.

CHAIR CORRADINI: From the ground up?

MR. UPTON: Yes.

CHAIR CORRADINI: Okay.

MR. TUCKER: Well, if you remember, it was horizontal versus vertical.

CHAIR CORRADINI: That's right. That's the major difference I thought that I remembered.

MR. UPTON: Well, even the configuration of the tube bundles. I mean, we've got two separate modules off of the steam head, which is different than the configuration like in the early BWR-2s, 3s that have isolation condensers.

MR. KRESS: I thought those tests were to check its heat transfer properties.

MR. DEAVER: That's all they were for.

MR. UPTON: That's true. It was the heat transfer properties.

MR. KRESS: I don't know why that addresses the materials question.

MR. UPTON: Just that the prototype was designed with the material that we're talking about, the low alloy inconel.

MR. KRESS: But they weren't given--

MR. UPTON: It was not. No, no. It was not. It was given the --

MR. TUCKER: If we were giving you a wrong impression that we were trying to do some special material testing. It was just built with the same material that we're going to use, and it was full scale to demonstrate that our design would work. And that with the vertical arrangement that we have better gas, inert gas relieving and it's --

CHAIR CORRADINI: Better than the horizontal?

MR. TUCKER: Yes. Yes, sir.

CHAIR CORRADINI: That's what I wanted to eventually get to. So you understand the --

MR. TUCKER: That difference in arrangement--

CHAIR CORRADINI: -- bundle because of that in the older designs you had certain of the higher tubes essentially less than efficient because of the accumulation of the noncondensers?

MR. UPTON: That's correct. That's correct. That the heat transfer characteristics for the upper tubes was reduced. And so what we have now in the current configuration is that we've got a vent of noncondensibles continuously. So the heat --

CHAIR CORRADINI: That's that additional line that's coming down through?

MR. UPTON: Yes, it goes to the new steam lines.

MR. DEAVER: Yes. Exactly.

MR. UPTON: That's correct.

MEMBER ARMIJO: So is the isolation

condenser always hot?

MR. UPTON: Yes, it is always hot. In other words, it's solid up to above the pools. The steamlines are always open. Okay. And then once you get condensation and the tubes themselves are filled, until you open the injection valves. And that will drain the IC and begin its operation.

CHAIR CORRADINI: So if is the right place. If you want to postpone us to a later time, that's perfectly fine, too.

So is that when you say "hot," that's not exactly completely correct. Because once you fill those tubes and you've got water all the way from essentially the black valves back up --

MR. DEAVER: Right.

CHAIR CORRADINI: -- you've got a cold leg and cold all the way up through the tube bank. And you're hot through the leak. And then you're essentially in a gradient condition below that if I understand how this thing operates, right?

MR. UPTON: That's correct.

MR. DEAVER: Yes. We fill it all the way to the top --

CHAIR CORRADINI: Well, yes. I was going to say, it essentially accumulates?

MR. DEAVER: Yes.

CHAIR CORRADINI: It is filling up.

MR. DEAVER: And that's important for once we open up the line we need that water in the drain.

CHAIR CORRADINI: No, no. Right. That I was clear on on the previous presentation you guys gave.

But I'm curious about this when you start up. So you said you tested these. Did you test up in a start up condition where you essentially filled them--

MR. UPTON: Yes.

CHAIR CORRADINI: -- drained them and looked at the associated transient of the drain?

MR. UPTON: Yes. Yes, that was done.

CHAIR CORRADINI: Okay. And that's been in the document --

MR. UPTON: You're worried about the gradient -- yes

CHAIR CORRADINI: I can go find it somewhere?

MR. UPTON: That's correct.

CHAIR CORRADINI: Okay.

MR. MELITO: Just to address your question about material conditions and what we've done, I think two things that we ought to note with this design versus the older design is:

(1) The vent for the older design was essentially out in the building itself because it was designed that way. And it was an air operated valve, in many cases, which sometimes was not open. So you would have problems with the zone design accumulating and nonconsensibles not being properly vented or being poorly vented. In this design it's a much more robust reliable design to keep it open and keep it vented.

The other thing about it is it's not sitting in a stagnant tank of water that's filled wit de-minned water. We actually have a cleanup system that runs separately for these pools to keep that water quality high. So we don't have the problems that you might have with a shell and tube arrangement in the older designs where the water was just sitting in there stagnant and maybe didn't turn over very often.

MR. DEAVER: Yes. That's part of the fuel--

CHAIR CORRADINI: What is the calculated heat leak then since it's hot or kind of hot, almost hot?

MR. UPTON: I'm not sure I understand your question.

CHAIR CORRADINI: Okay. If it's hot, that means you've got some thermal heat going that way all the time. What is it? A tenth of a percent? Do you know what your heat leak is through your isolation condenser at full power?

MR. UPTON: I don't have that number, but I'm sure it's been calculated. Because we had to worry about the -- that's lost from the containment through the system.

MEMBER ARMIJO: Right. But if you wanted to do materials testing on the tubes for the isolation condenser, what temperature would you pick? Would it be the coolant temperature or the water temperate? What temperature do they read? What's the steady state operating temperature of the isolation condenser, I guess that's what I'm asking?

MR. UPTON: It would be the pool temperature during normal operations. It's not normally used, so that's --it's kept below -- it's 110. I mean --

MEMBER ARMIJO: Really cold?

MR. UPTON: It's cold, yes.

MR. DEAVER: It's going to be closer to the cold side than the vessel hot side.

MR. UPTON: Right. That's correct.

MEMBER ARMIJO: Okay.

MEMBER ABDEL-KHALIK: You said something that the water is not allowed to accumulate?

MR. DEAVER: Well, it is allowed. Yes.

MEMBER ABDEL-KHALIK: Is allowed?

MR. DEAVER: Oh, yes.

MEMBER ABDEL-KHALIK: Okay.

MR. DEAVER: Yes. The idea is to back the water up all the way into the system. As a matter of fact, we added --

MEMBER ABDEL-KHALIK: I was just trying to clarify that.

MR. DEAVER: -- an extra to add more capacity. Yes. Okay.

MEMBER ABDEL-KHALIK: Okay. Thank you very much.

MEMBER ARMIJO: Okay. Now I understand it.

MR. DEAVER: Okay. As far as main steamline and feedwater piping, this arrangement is a plan view of --

MEMBER ABDEL-KHALIK: If we go back to the previous slide --

MR. DEAVER: Okay.

MEMBER ABDEL-KHALIK: -- you're talking about the safety and relief valves.

MR. DEAVER: Yes.

MEMBER ABDEL-KHALIK: Is there anything new in the design of these SRVs?

MR. DEAVER: Right now on the SRVs we actually identified two types of valves. One is the direct acting, which has been the typical valve. But we also have identified a pilot operated valve as a potential valve also. Joel is more familiar with that.

But we have been trying to consider the advances in valve technology of recognizing that the MISVs or SRVs have had a lot of maintenance issues in the past, we have been trying to evaluate the new technologies that seem to work and have been used in similar applications.

MEMBER ABDEL-KHALIK: What is the experience base of direct versus pilot operated valves?

MR. DEAVER: You want to answer that?

MR. MELITO: The experience base is pretty much from the BWR-5s forward they have been using spring closed direct acting valves. The earlier plants relied on a piloted valve, which was a depressurized to operate type pilot valve, very typical of piloted valves. And it was only of one particular design basically, even though there were a couple of variations of it. It was still basically one valve.

Because of the history with that valve, there's not been a lot of effort to look at piloted valves until we looked at it more recently to see if there were updates in design and see if there was a better way of putting together a piloted valve that would not have the problems that the older valve experienced.

So right now there is in the world another valve design that is a piloted valve and that has been primarily used in PWRs outside the U.S. It's got limited application in boiling plants. And so we need to look at it very intensely before we make the final decision as to whether or not we want to use that valve or stick with the valve that everybody knows and pretty well from history the direct acting spring closed valve.

MEMBER ABDEL-KHALIK: So these pilot valves are totally different than the old style pilot valves --

MR. MELITO: Yes, they're very different.

MEMBER ABDEL-KHALIK: -- with which a lot of people have had problems?

MR. MELITO: Right. And, you know, a lot of people compare the two valves, the traditional piloted valve and the spring closed valves where the spring closed valve is termed direct acting and the pilot valve is then termed reverse acting. In the nature of the way they redesigned this newer concept of piloted valve, it's essentially a direct acting piloted valve, which sounds like an oxymoron, but in looking at the details of the value, it's how you would have to term it.

What they've done is to use a principle that was developed by the Europeans after TMI-2 accident for their piloted values, which eliminates having a pressurized piston chamber that gives you the high potential for an inadvertent valve opening because of problems with the pilots.

This design relies instead on keeping all the steam isolated below the pilot seat as close to the steamline as possible and keeping the pilot shut as close again to the steamline as possible. And then opening the pilot to pressurize the piston chamber and open the valve only when it's called upon. And it divorces the functional requirements for the valve so that there's only a mechanical pressure lift for safety mode function, and there's a totally separate pilot that is a solenoid-activated pilot for the relief mode functions so that those two functions no longer overlap.

MEMBER ABDEL-KHALIK: Would the direct valves also have that capability of being remotely operated from the control spring?

MR. MELITO: The spring closed valves are equipped, in particular for the SRVs. And that's the distinguishing feature between an SRV and an SV. The SVs are basically all the same,they're all spring lift. To make it an SRV, what's done with the spring closed valve is to add an actuation mechanism that uses a pneumatic cylinder that is pressurized up to lift a lever arm and cause the valve to be lifted open. It basically overcomes the seating force of the spring in that mode. So it can operate over a wide range of pressures.

And all of those valves using that pneumatic design are designed with a backup accumulator. So there's always accumulator service to operate those valves.

MEMBER ABDEL-KHALIK: Thank you.

MR. DEAVER: Okay. I'll just mention, the last item is component supports. And we simply design to ASME codes, subsection MF for all the support design. So there's nothing new or different in that area.

MEMBER ARMIJO: But this will be the heaviest vessel you've ever supported?

MR. DEAVER: Yes. Yes.

MEMBER ARMIJO: Plus all the other stuff, the chimney and --

MR. DEAVER: Right. Yes. We have a little different design there. In the past we've had a skirt that the issues there related to temperature gradients going down through the skirt into the concrete foundation. And we have an alternate design that basically accommodates the thermal expansion without getting into those temperature gradient issues.

MEMBER ARMIJO: Okay.

MR. DEAVER: Okay. This is the arrangement of the main steamlines and feedwater lines. Basically the four steam lines simply come off the vessel, curve around and go through the containment penetration in a very typical manner that we've had in the past. And you can see the SRVs and the SVs located on the lines.

And then feedwater, we have basically six nozzles that enter/flow into the vessel. Each of those are channeled into one header. And so we have two lines that exit the vessel, and they happen to be on the outer side of the steamlines and a little higher elevation.

Okay. Well, the last slide on the section description is the DPV valve. This basically shows the unfired and fired position. Basically in this design, this is the inlet or the side that sees the vessel pressure. And this member is a continuous member that the very operation sealed because it's a continuous membrane.

MEMBER ARMIJO: What holds it against the pressure? What's holding that thing from flipping open?

MR. DEAVER: It's one piece.

MEMBER ARMIJO: It's one piece?

MR. DEAVER: Yes, it's a one piece design. It's a --

MEMBER SHACK: It's solid metal.

MR. DEAVER: It's solid metal.

MEMBER ARMIJO: Okay. That'll do it. Okay. I understand it now. I was wondering what was holding it on.

MR. DEAVER: Okay. I guess I wanted to

point out that this is what we term a squib valve. And this is a type valve that we've used in the past in the standby liquid control system on BWR-5s and 6s and so forth. So we have quite a bit of operating history with this type valve.

It basically uses an actuator here that's ignited that shears the pin and drops the cylinder. And then shears this cap off and then it opens up.

This is what we call a passive valve, but once it's opened, it's opened. You'd have to go in and service it before you could close it up again.

MEMBER ABDEL-KHALIK: But there's no way to test this valve?

MEMBER ARMIJO: No.

MEMBER ABDEL-KHALIK: To test the adequacy of the squib?

MR. DEAVER: In operation? No.

Well, the program for this is to take these booster charges and in a five year period basically take a sample, take them off and actuate them and demonstrate it that they're functioning.

CHAIR CORRADINI: But you had said at the beginning you have experience with these valves. I forgot what you said is the --

MR. DEAVER: The standby control.

CHAIR CORRADINI: Okay. Thank you.

MR. DEAVER: Yes. That system has used these for quite a long time. And it had positive operating experience with those.

MEMBER SHACK: You haven't used it?

MR. DEAVER: Pardon me? Well, they have been used a couple of times.

MEMBER SHACK: Inadvertently.

MR. DEAVER: And it did work, yes.

MEMBER SHACK: Yes.

MEMBER ARMIJO: If that tension bolt fails when you don't -- then you'd have a depressurization?

MR. DEAVER: Yes, you would.

MEMBER ARMIJO: I imagine there's lots of margin in the tension that exists?

MR. DEAVER: Yes.

Joel, do you have something else?

MR. MELITO: Yes. I was going to say the margin is enough because of the purpose of that tension bolt in the design of this valve is that the propellant in the booster has to burn enough to generate that high pressure gas charge so that when it is enough to break that tension bolt, it's going to give you a quick knife action to shear that cap off. It can't be a low press. It's got to be a real snap action to make it work effectively.

So the purpose of the tension bolt is to hold it until that gas charge is built. So it's got a lot of margin.

MR. UPTON: Jerry, this is Hugh again.

You might also mention the full scale test facility of the valve at Wylie Labs. So we have developed a prototype and it has been tested.

MR. DEAVER: Yes. That was done in the early '90s as a part of the SBWR program. Basically the same valve size and everything is planned to be used on the ESBWR.

MEMBER ARMIJO: But these are bigger valves than the prior standby liquid control valves?

MR. DEAVER: Yes. Yes. They were typically two inch variety. This the inlet diameter is more like eight inches.

MEMBER ARMIJO: A big valve?

MR. DEAVER: Yes.

MEMBER ARMIJO: Okay. And tested?

MR. DEAVER: Right. Any more questions on that?

MEMBER MAYNARD: Is the charge the same size as what --

MR. DEAVER: No, they'll ge larger.

MEMBER MAYNARD: Okay. Is there experience with this size of charge for the squib? I understand the valve is bigger, you've tested it.

MR. DEAVER: It was tested, yes.

MEMBER MAYNARD: We've had a lot of operating experience with the smaller squid valves with the smaller charge.

MR. DEAVER: Right.

MEMBER MAYNARD: I'm wondering is this a charge that has been used in the industry or is this something new that we're having to get the same design but a bigger one that we don't have history with?

MR. DEAVER: Well, we did test it.

MR. MELITO: This is Joel Melito. Let me answer that question.

Essentially if you're driving a late model car, it's there in front of you. It's the same thing. It's the same kind of stuff --

CHAIR CORRADINI: So that gives you confidence?

MR. MELITO: No. Actually I've had mine go off. It's not so much of an air bag as a hot gas bag if you've ever experienced one.

MR. KRESS: Yes. They burn, yes.

MR. MELITO: But, yes, these are used in

air bags. They're used in ejection seats. They're used in helicopter floats if helicopters crash in the sea. There's quite a lot of experience with these things, most of it outside nuclear industry.

MEMBER MAYNARD: And I understand with these types. I'm just wondering if there's anything unique about this particular chart. This is the same type of -- you said it's a bigger charge than what you've had in the standby liquid control systems.

MR. MELITO: Yes.

MEMBER MAYNARD: Is it, though, something that there is a lot of experience with this size of charge on it? Okay. That's what I'm asking.

MR. MELITO: Yes.

MR. DEAVER: I might add that there's not just one charges, there's actually -- to get the N minus two criteria we have actually four charges on it.

MR. MELITO: Just to clarify that, there are four ignitors for one booster charge.

MR. DEAVER: Yes.

MEMBER ABDEL-KHALIK: Now there have been some experience with Bell helicopters where the manufacturer recalled all these squib valves but essentially they had the wrong squibs.

Now you are not planning to build your own valves, are you?

MR. DEAVER: Right.

MEMBER ABDEL-KHALIK: So how do you QA these squibs?

MR. DEAVER: Well, there will be batches and we'll be testing samples from the batch.

And as a matter of fact if in actual operation we tested one that failed, we would take all the charges out of that batch out of service is the plan.

MEMBER MAYNARD: But when you get a new batch in, do you test a sample of those?

MR. DEAVER: Well, as part of the acceptance of those, we would, yes.

MEMBER MAYNARD: Okay. That's what I thought.

MR. KRESS: Does the charge age?

MR. DEAVER: Pardon?

MR. KRESS: So every now and then you have to replace them?

MR. MELITO: We know what a minimum life is right now. We don't know a maximum life. So the plan going forward is to at least adhere to the minimum life and then determine from service whether or not there's a longer life that's suitable. Customers would obviously be interested in that from a cost standpoint.

MR. KRESS: Yes.

MR. DEAVER: But the initial program is on a five year time scale.

MR. KRESS: How are they ignited? Is that something like a spark plug in there?

MR. MELITO: No. The ignitor is really a different kind of fast reacting pyrotechnic charge. It sets off more easily from a current. There is basically a set of wires. And if you see the little stub sticking out the lower one of the two wires, there'll be four like that for four divisions of ignition.

MR. KRESS: Yes.

MR. MELITO: And each of those two wires going in has at least two bridge wires. So even if one of the bridge wires fails for some reason, there's another one that can set off that particular charge.

MR. KRESS: So a quick current there heats it up really fast?

MR. MELITO: Right. Heats it up, sets off that charge. That charge then creates enough heat to set off the main booster charge.

MR. KRESS: What provides the current? Do you have a battery?

MR. MELITO: Yes, it's all battery backed up, all four divisions.

CHAIR CORRADINI: Either the DC, this is the battery system.

MR. DEAVER: Yes. Safety related battery supplies.

MEMBER ARMIJO: What happens if there's a spurious actuation or something just fires off one or more of these boosters? You got a -- you got a like plan and it's more than a simmer? You've analyzed, I'm not sure. But, you know --

MR. DEAVER: Well, going back into the instrumentation control systems, that's where they're going to have triply redundant logics and such that, you know, which are basically foolproof. You know, for spurious lightening strikes or pulses or anything that might try to actuate it. So you'd have to get more than one signal to actually actuate it. You know, it has to be confirmed before it would actually actuate. So there's a very solid logic path --

MEMBER ARMIJO: Right. But let's assume all of that failed and you actuated one of these when you didn't want to, is it a big deal or a small --

MR. DEAVER: Well, it's basically going to blow down the reactor.

MEMBER ARMIJO: That's it?

MEMBER SHACK: It's going to blow up the rest of them quite shortly thereafter.

MEMBER ABDEL-KHALIK: What is the combined total area of these lines compared to a main steamline?

MR. MELITO: All eight?

MEMBER ABDEL-KHALIK: Right.

MR. MELITO: One is okay.

MEMBER ABDEL-KHALIK: If they're eight inch lines, then they're smaller than a main steamline.

MR. DEAVER: Oh, yes. Well I know the opening pipe coming into to it is eight inch versus the large steamlines.

MEMBER ABDEL-KHALIK: So they're bounded by main steamline break inside containment?

MR. DEAVER: Oh, yes. Yes.

MR. MELITO: I think inadvertent actuation of the DPV, it's going to be bounded by your main steamline break.

MR. DEAVER: That's our limiting design case right at the moment.

MR. MELITO: Yes. I'm getting about 400 square inches for all eight if all eight spuriously opened and a single steamline being more like 600 square inches. So you're still bounded.

Although now that I think about this again, you've got a 14 inch chokepoint and you won't have that if were to get into the event of all eight going off.

MEMBER ARMIJO: But if one goes off, you might as well fire off the others.

MEMBER SHACK: But we analyze it in 15.3.14

MEMBER ARMIJO: Okay. We'll have another shot at it.

MR. KRESS: Yes. Yes. But the question I'd have is is there frequency in there --

MEMBER ABDEL-KHALIK: Is that really true with the nozzle limit, throat limited to 14 inches on the steamline?

MR. MELITO: Yes. The venturi is a 14 inch throat.

MEMBER ABDEL-KHALIK: No. I mean the -- the opening is involved --

MEMBER ABDEL-KHALIK: How long would it take before the --

MR. MELITO: -- and the failure of all these valves would be limited to --

MEMBER SHACK: It's ten to the minus --

MEMBER ABDEL-KHALIK: -- were bounded by an end steamline break that would --

CHAIR CORRADINI: Yes, but you're not -- but it's not to the flow rate. It's essentially the enthalpy, the power you're discharging. So your enthalpy that you'd would put it on in the steamline is going to be held a lot different than the water you're going to take out of this, unless this is expected to be a steam discharge out of this?

MR. MELITO: This is a steam discharge. It is very close to the steamline elevation.

CHAIR CORRADINI: Oh, I thought it was lower than that?

MR. DEAVER: Just slightly lower. But not very much.

MEMBER ARMIJO: Steamline LOCA.

CHAIR CORRADINI: Okay. Any other questions from the Committee?

I'm sorry, did you have-- I thought this was your last one.

MR. DEAVER: Well, just a summary. And that is that we're basically using proven technology even for the new systems or technology that has been successful in the past. And we're currently working with the Staff to resolve issues. I think we have a lot of them that are basically in process that we have a understanding but we haven't implemented or responded to RAIs. So I think we're on a path that we're success at this point.

CHAIR CORRADINI: Questions by the Committee?

Should we take a break?

MEMBER SHACK: Yes.

CHAIR CORRADINI: How about five after 2:00? We'll have the Staff back up ready and willing.

(Whereupon, at 1:52 p.m. a recess until 2:05 p.m.)

CHAIRMAN CORRADINI: We have a new team. Is it Eric? Are you going to start us off, Eric?

MR. OESTERLE: Yes.

CHAIRMAN CORRADINI: Okay.

MR. OESTERLE: Well, thank you, everyone. Thanks for staying. My name is Eric Oesterle. I'm a Project Manager with the Office of New Reactors, Division of New Reactor Licensing. I was the lead PM for the Chapter 5 ESBWR review, but that doesn't mean I did the review. These gentlemen here who are the experts did the review, and they definitely had some help.

The purpose of this afternoon's presentation is to brief the Subcommittee on the staff's review of Revision 3 of the ESBWR design certification application, specifically Chapter 5, reactor coolant system and connected systems. And we are also here to answer the Committee's questions regarding the staff's review.

As you know, we have received Revision 4 of the ESBWR design certification application, and it is currently undergoing staff review. The results of the review may resolve some of the open items that the staff identified and documented in the open item letter, which you all received and that are also included in the safety evaluation report with open items.

They may be resolved by Revision 4 or later by Revision 5, and, in addition, there may be some new requests for additional information that arise as a result of the staff's review of Revision 4.

This is the team that was assembled for the review of Chapter 5. The lead technical reviewers were John Wu, and I will be going over his review of Section 5.2.1, George Thomas to my right will go through the identified sections, Robert Davis, next to George, will go through 5.2.3 and 5.2.4, and then we have Chang Li and Neil Ray will go through their respective sections as well.

We also had some secondary reviewers that provided input to these sections, and that included Yamir, Lambrose, John Fair, and John Huang.

As in the other reviews, the way we formatted this presentation is to talk about the applicable regulations that the staff relied on to do their review, an RAI status summary, selected SER technical topics, a discussion of some of the significant open items that the staff identified in the review, a discussion of some of the COL action items, and of course we welcome any questions that the Committee has during our presentation at any time.

On this slide is a list of the regulations and review guidance that the staff relied upon to perform the review. Subpart B of Part 52 on standard design certifications, there are various applicable sections from Part 50, including the appendices listed there. There were numerous general design criteria that the staff relied upon to do the review, and the regulatory guides and the SRPs listed there, in addition to some other guidance which included generic communications, NUREGs, and SECY papers.

As far as the RAI status summary goes, we had a total of 138 requests for additional information originally. Of those 138, 118 of them have been resolved so far. Of those, there are approximately 20 that remain as open items, and that number may be within about one or two accuracy, depending upon the updates that we have received since we issued the SER with open items and the open item letter. Some of the significant open items will be discussed later on some of the section discussions.

This is just a list of some of the subsections that we will be going through, 5.2 on the integrity of the reactor coolant pressure boundary; 5.3, the reactor vessel; 5.4, component and subsystem design. And if you're looking at the numbering, it looks like there are some sections that were missed. Those sections are applicable to pressurized water reactors and not to the ESBWR, so they were not included.

For Section 5.2.1, I am presenting the review that was performed by John Wu, and this was on compliance with ASME codes and standards. The staff performed a review of the ESBWR design and determined that the design will comply with all -- with the ASME code section requirements and the applicable code cases.

Next up we have George Thomas to give us a summary of his review of Section 5.2.2 on overpressure protection.

MR. THOMAS: My name is George Thomas. I am from the Reactor Systems Branch. We reviewed the system according to the standard review plan 5.2.2, and basically our main questions we had on the SRP setpoint drift, seal decay, and the summary.

They told us that they did not finalize the design yet of the SRP, so in -- so that they finalize the design, then they will consider all of these issues. And they will be meeting the ASME section 3 and 11 requirements.

And we also performed an overpressure analysis using Part D for the equilibrium code. And the peak calculated pressure was 1,263 psig, which is below the limit of 1,375 psig.

We got a COL action item, so we will be doing analysis again for the initial code.

MEMBER ABDEL-KHALIK: I guess there were several questions as to whether the limit is 120 percent or 110 percent.

MR. THOMAS: 110 percent for the

overpressure analysis. So in Revision 4 the correct --

MEMBER ABDEL-KHALIK: Okay. Because what I have here is Rev 3.

MR. THOMAS: No.

MEMBER ABDEL-KHALIK: And it said 120 percent.

MR. THOMAS: No, that is for ATWS. See, ATWS the criteria is 120 percentage, and the ASME is 110 percentage.

MEMBER ABDEL-KHALIK: Okay.

MR. THOMAS: Okay. You know, the ASME level --

MEMBER ABDEL-KHALIK: Right.

MR. THOMAS: -- B and C, you know, that is different.

MR. OESTERLE: Next section is Section 5.2.3 on reactor coolant pressure boundary materials, and that will be Bob Davis.

MR. DAVIS: I'm Bob Davis, and I am in the Component Integrity Branch in the Division of Engineering in NRO. And I reviewed the reactor coolant pressure boundary materials and used standard review plan 5.2.3 as a guide during that review.

With the exception of satisfactory

resolution of the open items that I will talk about later, the reactor coolant pressure boundary materials are found to comply with the requirements of ASME Code Section 3. All ferritic materials in the reactor coolant pressure boundary conform to Section 3. Ferritic materials, piping, components, bolting, meet the fracture toughness requirements of ASME Code Section 3.

For the austenitic materials used in the reactor coolant pressure boundary, all austenitic stainless steel are supplied in the solution heat treated condition, and those materials to be welded have a carbon content of less than .02 percent, which is consistent with the NUREG-0313 technical report on material and processing guidelines for BWR coolant pressure boundary or piping. And they are also in compliance with Reg. Guide 1.44 on control of the use of sensitized stainless steel.

The cleanness and cleanliness requirements conform to Reg. Guide 1.37 to ensure that there are no contaminants that may promote intergranular stress curves and cracking or other forms of degradation. And the reactor coolant pressure boundary materials are compatible with the reactor coolant water chemistry, which is -- which will be maintained in accordance with Reg. Guide 1.56 and the EPRI series, BWR water chemistry guidelines.

In the following --

MEMBER ARMIJO: The staff doesn't have any concern about post-weld grinding of stainless steel welds?

MR. DAVIS: Well, they cover that in their DCD, and they also cover it in RAI responses, where they have said that they will control grinding. And then, when grinding is performed, they will polish those areas to ensure that they don't maintain residual stresses.

And there is a lot of controls for maintaining that. If you are going to weld large components, it is going to be impossible. It is actually impossible that you are not going to grind on those components somewhere.

MEMBER ARMIJO: I disagree, but the -- there is no way to tell that you haven't got residual stresses at the surface of those ground materials. They are -- will initiate and you have test data to show it -- IGSCC. And if it initiates, it will propagate. So the issue is: is the staff being a little too tolerant about those weld grinding?

MR. DAVIS: Well, I think that in one of

their RAI responses, for example, they apply special cold work controls to all stainless steels. They have hardness requirements of no greater than Rockwell 90B.

MEMBER ARMIJO: Yes, but that -- you know, if you've got a 10 mil surface hardened layer, the Rockwell hardness on the bulk will give you the bulk properties, not the surface properties. So a lot of these tests are kind of illusions that you have solved the problem.

MR. DAVIS: Well, I think other than controlling another -- applying the controls that -- the controls that they have specified that they are going to use, that they say that they have qualified through procedures, like when they grind they have specific procedures if they grind on a material, I'm really not aware of what could be done beyond that. I mean, I could certainly consult with --

MEMBER ARMIJO: I think that the --

MR. DAVIS: -- the rest of the staff, and I could look into, but --

MEMBER ARMIJO: I think you should.

MR. DAVIS: Right. But if you are going to weld large components, I don't see how it is going to be performed without grinding somewhere, because there is no such thing as a perfect weld. And it will always -- something will always have to be repaired somewhere.

MEMBER ARMIJO: I think that is entirely too tolerant. It is not a given that it is impossible to make good welds the first time.

MR. DAVIS: That is certainly the idea. But, I mean, in reality it is extremely difficult.

MEMBER ARMIJO: Okay.

MR. DAVIS: Okay. The significant open items -- GE-H was missing some material specifications for some of the classwater/feedwater check valves. The earlier open item for the use of ASTM 800 they have already resolved, because they intend to use Hull's Equivalent Factors. There are some filler metal specifications that they need to correct in their DCD.

And the justification for using ASTM A709 HPS70, our issue isn't that they're using that material for structures, because first off that will be reviewed as part of Chapter 3. That material is allowed for use by Supplement 2 and 690 to be used for internal structures. The issue is joining that material to the containment liner without doing a post-weld heat treat.

HPS70 is a quotient-tempered steel, which means that post-weld heating treating it and diminishes its mechanical properties, yet the code requires when you weld I think over an inch and a half thick materials on the containment liners it requires a post-weld heat treat. So we are still trying to resolve that issue with them, and we are actively engaged with this code case that they have presented.

MEMBER SHACK: And what is the code case going to say?

MR. DAVIS: Well, the code case is basically going to say -- it is going to allow attaching this material to the liner without post-weld heat treat, and it has a lot of additional requirements for procedure qualification testing, toughness testing, and those things.

I think that what our concern is is that we perfectly understand why you don't want to post-weld heat treat a quotient-tempered steel. The question is, is why is it okay to not post-weld heat treat the liner material? This material welded to itself is -- it's a very weldable material. It was developed between the Department of Transportation, the Navy, and industry.

MEMBER SHACK: I looked for it on Google, and it seems to be used in every bridge specification now. I mean, it is standard bridge material.

MR. DAVIS: It is designed to be -- you know, most of the time they design the material and then try to figure out how to weld it. This material was designed with the intent of welding with reduced preheats.

And with the low -- not gaining its strength from carbon, so that it is a lot easier to weld, and it is -- I think they have used it to fabricate over 200 bridges for, what do they call it, the fracture-critical members on bridges and it is approved for use in all areas of the United States, no matter what the temperature range is. It makes it extremely tough.

MEMBER ABDEL-KHALIK: The missing material specification for the feedwater check valves, is that inadvertent, or is there a real problem with regard to historical performance of feedwater check valves?

MR. DAVIS: No. It's just that it is a major -- it would be considered somewhat of a major component, and it is not there. I mean, I could only assume that it will be 2-1/4 -- some specification that will be 2-1/4 prone, since the feedwater lines are 2-1/4 prone. But it's --

MR. DEAVER: This is Jerry Deaver again.

I just wanted to mention that that was just an inadvertent omission. It wasn't intentional in the certification document.

MEMBER ABDEL-KHALIK: Okay. Thank you.

MR. DAVIS: And now I'll move on to -- I guess one note on the HPS steel, that is a weathering steel. I don't know if that was brought to your attention before. It is a -- that's --

MEMBER SHACK: Core 10.

MR. DAVIS: It's -- well, it's not Core 10. There are a lot of different types of Core 10, but it is similar. It has a lot of copper in it, and it forms a copper oxide layer.

I am going to move on to Section 5.2.4, which is pre-service and in-service inspection of reactor coolant pressure boundary. With the exception of open items previously identified, PSI and ISI reactor coolant pressure boundary was found to comply with the requirements of 10 CFR 50.55a and ASME Code Section 11.

Development of the pre-service and in-service inspection programs is the responsibility of the COL holder. Obviously, the COL applicant can't come forth with a whole program, because a plant hasn't even been built yet. That will be done at a later point.

The pre-service inspection and in-service inspection ultrasonic examinations will be performed in accordance with Section 11, including the conditions in 10 CFR 50.55a, meaning Appendix A, performance demonstration, initial qualified exams.

All items within the Class 1 boundary are designed to provide access to perform pre-service inspection and in-service inspection examinations required by IWB-2500.

For piping, pumps, valves, and supports, welds are designed to permit ultrasonic examination from at least one side. Where component geometries permit, access from both sides is provided. GE-H has indicated that radiography may be used for pre-service inspection and in-service inspection. The staff is concerned with this approach, because current operating plants often seek relief from performing radiography due to ALARA and other issues.

MEMBER ARMIJO: Why are you concerned? I mean, as long as you get a good volumetric inspection, why do you care?

MR. DAVIS: Well, we're concerned because on austenitic to -- or austenitic weld metal, which would be an austenitic to austenitic weld or a dissimilar metal weld, if you don't have access from both sides, you cannot do a PDI, performance demonstration initiative, qualified UT exam, which means that the only exam you could do -- you would have to -- like say if you could get the one side of it but not the other, you would have to supplement it with radiography.

But the problem is, in reality, in operating plants licensees do not like to do radiography. The pipe might be full of water, everybody has got to get out of the containment. So our concern is is that if -- if a weld is planned -- you know, when it is designed, they say, okay, part of this can receive a radiography -- a radiograph. Then, down the road a licensee will come in for relief from doing that examination.

So all welds have to be designed. You can't design a weld that is impractical to inspect. So that's our concern, and we are actively engaged with GE-H to work something out with that.

MEMBER SHACK: Is there going to be a pre-service baseline ultrasonic exam?

MR. DAVIS: Well, all of the Class I gets pre-service over a certain diameter, and then -- and the vessel is completely inspectable, you know, 100 percent UT for pre-service. After it goes into service, there are some limitations on some of the nozzle welds. But even though there are some limitations, they can still do those inspections in accordance with a staff-approved code case.

Once we go move into some of the other class -- the other Class 1 welds, not the reactor pressure vessel, we are --

MEMBER SHACK: I was just concerned -- it's just always easier to interpret the later inspection if you knew what it looked like --

MR. DAVIS: Yes.

MEMBER SHACK: -- the first time.

MR. DAVIS: Well, everything Class 1 has to get a pre-service over a certain diameter.

MEMBER SHACK: But there's a pre-service volumetric.

MR. DAVIS: Pre-service -- well, it has to use whatever method will be used later on is what it will have to use. Now, when you go to -- for Class 2, that will be -- I think it's 25 -- there's a smaller percentage that has to be done, and those welds or what welds get inspected will be designed -- decided by the COL holder. The owner is going to decide which ones they want to do a pre-service inspection on.

But whether they do combination UT/RT or RT on the pre-service, our concern is is somebody asking for relief later on, because now they claim, oh, it's impractical to do RT because of ALARA or some other issue.

Any other questions on that?

(No response.)

And this leads to the open items, which are similar to the issues we just discussed. The ISI of austenitic and dissimilar metal welds and proposed use of radiography is one of the major open items on this section.

From what I understand, this issue has been resolved, but just not in writing yet. In the DCD, in one instance they say that they can perform the nozzle examinations using an NRC-approved code case, but yet in another section they discuss about the possible need for relief. And from what I understand talking to GE-H they can do these inspections, and they will remove the part that discusses possibly asking for relief for the nozzle inspections.

We have requested GE-H to modify the DCD to include a COL action item for the COL applicant to fully describe their PSI and ISI programs, and augmented inspection programs, FAC, and I think there is one other for between containment isolation valves, that even though they can't supply us with their full ISI program and PSI program, they can provide a pretty full description of everything that it will include when they submit their application.

And that is the only -- the only COL action item currently is that the COL holder is responsible for the development and implementation of PSI and ISI program plans that are based on ASME codes.

MR. OESTERLE: Okay. Next up is SER Section 5.2.5, and Chang Li will discuss reactor coolant pressure boundary leakage detection.

MR. LI: My name is Chang Li. I am with Balance of Plant Systems Branch. I reviewed the reactor coolant pressure boundary leakage detection.

My review was based on standard review plan Section 5.2.5, which refers to Reg. Guide 1.45, Revision 0, and based on some operating experience, specifically the operating experience for the Davis-Besse event that low-level leakage, even far below tech spec limit, if lasted for a long time the material degradation and the stress corrosion could be a safety concern.

In Davis-Besse, they recognized that corrosion resulted in the reactor vessel wall reached to a dangerous thickness. Taking prompt corrective action by plant operator is the key to avoid occurrence of such events. To the first COL holder item, which is in the third bullet there, is an operating procedure that needs to be developed by the COL holders.

The GE-H design of the alarm setpoint of five gpm for the unidentified leakage, which is the same value as the tech spec limit of five gpm, that needs to be lowered to support the operating procedures I just described, which provide an early warning to the operators and lead the operator to take necessary measures.

This is the open item 5.2 --

MEMBER SHACK: Okay. So you don't have any problem with a tech spec for an unidentified leakage of five gpm. You just want an alarm at a lower level.

MR. LI: Yes. Yes. We review -- starting with the question about both tech spec limits, as well as alarms, after we discuss with GE about the basis -- they have been telling us how they established the tech spec limit. We agree with them in terms of the tech spec limit. However, we believe that the alarm limit needs to be lowered in order to give the operator early warning instead of giving the alarm at the point that they need to shut the plant down.

MEMBER ARMIJO: We heard earlier from GE-H that, because of the design and a lot of the -- a lot of water in the containment that they can't do 1 gpm.

MR. LI: They can't do one, so --

MEMBER ARMIJO: the question is: between one and five, what --

MR. LI: Yes. We have been discussing to the point that it is going to be between one and five, and GE is going to develop the basis what's the -- there is a background leakage, and some delta that is set for alarm. So it will be somewhere between one and five, but --

CHAIRMAN CORRADINI: Can I just investigate a bit? So if I have an alarm below where they can reliably -- well, let me make sure I -- I mean, let me restate it, because I -- Sam said it maybe better. I don't understand. Is five or they are -- they are highly reliable, they can determine it, or --

MEMBER SHACK: It's a tech spec limit. When you hit five, it better be --

CHAIRMAN CORRADINI: So what concerns me is if I set an alarm below that, I will just ignore it, if I keep on getting a false -- if I keep on getting an alarm that I can't verify where it is, then the natural response would be just to blow it off as time marches on. So I'm trying to get an idea of -- it can't be one, so is -- so you're leaving it to the licensee to come up what it is, and then you will check it again?

MR. LI: Yes. We are looking for both coming from the -- from the applicants, and we are going to have to review, see if that's justified.

MEMBER SHACK: How about a delta over 24 hours?

MR. LI: That's the tech spec limit for earlier BWR plants, and we asked them and they believe that's -- that's coming from the IGSCC issues. And by this -- the ESBWR design, their material is upgraded and taken care of and --

MEMBER SHACK: No. But, I mean, it just seems to me that's a good action signal, rather than an absolute level -- the fact that, you know, it jumps so much over a certain amount of time.

MEMBER MAYNARD: Does it have to be an alarm, or can it be a methodology to calculate --

determine what your -- an unidentified leak rate is, and set administrative limits below five?

MR. LI: Sure. It could be, yes. But they haven't had anything, but -- since alarm is one way to trigger the operator action, and also in the standard -- in the -- in this Reg. Guide 1.45, there is a lot of criteria. So if they fix alarm criteria, fix the two departures that -- the problems that we identified in open items, of course they can.

CHAIRMAN CORRADINI: So let me just say it back to you one more time. You can see where I'm coming from, and then I -- I understand between you and the licensee. But if I have an alarm, then is that a way to start a root cause analysis? Because if it's not going to be in their tech spec, it would be some sort of -- I mean, the way I think is I'd start at five and I'd back off, and do some sort of, as Bill said, integrated measurement rather than alarm.

I just get concerned that once I have an alarm the natural response is -- when I'm uncertain anyway is I might not do the appropriate corrective action.

MR. LI: That's the third bullet, tells -- we ask the COL applicant to develop a procedure, and with the alarm to start -- they need to monitor training. And if you determine the leakage source, all those management processes --

CHAIRMAN CORRADINI: Okay.

MR. LI: -- need to be initiated. So that is one of the important COL action items we try to develop and ask GE to have it and start it at the level below tech spec limit.

CHAIRMAN CORRADINI: Okay. Thank you.

MEMBER ABDEL-KHALIK: The expectation is that they would do this every day?

MR. LI: No. It's only when it's triggered.

MEMBER ABDEL-KHALIK: No. I mean, as far as the trending.

MR. LI: This procedure works -- well, I think we are asking them to develop the procedure. How often, when it's every day or it's after it has been triggered that need operator attention, we don't -- we haven't specified in that level of detail with that COL holder to develop it.

MS. CUBBAGE: Chang, I might add -- this is Amy Cubbage -- that the tech spec surveillance requirement is every 12 hours.

MEMBER ABDEL-KHALIK: Twelve hours.

MR. LI: Yes.

MEMBER ABDEL-KHALIK: Thank you.

MR. LI: So that's different requirements there.

MEMBER ARMIJO: Well, since a BWR doesn't have boric acid, the -- you know, you clearly don't have the same threat. But you don't want to ignore these leaks. So I'm getting the impression that you are fairly flexible on what that --

MR. LI: Yes.

MEMBER ARMIJO: -- number is, as long as it is reliable.

MR. LI: Yes. And also, as well as the operator take action. Don't just ignore like a --

MEMBER ARMIJO: Wait until the tech spec --

MR. LI: Yes.

MEMBER SHACK: Yes. I think they shut Duane Arnold down at three and a half gallons.

MR. LI: Yes, they claim that a good operator is -- always do some actions, and there are procedures in place. But there is no standard, no requirements. So here we are having that -- all of the ESBWR are good operators.

MEMBER SHACK: Better.

MEMBER ARMIJO: Okay. That's fine.

MR. LI: So I just point out that the alarm setpoint issue, which actually one alarm -- by fixing that alarm usually it will fix the two open items, which is starting from the different angles. But it is all adding to having the -- having the plant operator do some action at very low leakage. How low it is, we will let the licensee to make that determination based on their operating experience.

The last bullet, COL holder --

MEMBER ABDEL-KHALIK: And you don't think that doing detailed fracture mechanics analysis on the size of the crack, and the propagation of the crack, consistent with the calculated or assumed or permitted leak rate is appropriate, that they can just do a simple back-of-the-envelope calculation on the size of the crack?

MR. LI: I think on that determination of five gpm, the way that they have analyzed and based on the experience of those previous BWR systems, and that number is -- seems to be conservative in terms of the structure, you know, mechanics that critical crack maybe grows to that.

So that number seems to be conservative enough. It is only the -- at the operating plant now that necessarily has to have operator actions when it's very low -- low leakage, way below tech spec. They can't ignore it. It just -- it's --

MEMBER ABDEL-KHALIK: Yes, it's --

MR. LI: And leakage within tech spec limit, there is nothing we need to do. So that's something we want to fix.

The last bullet under COL holder item is to convert different sources of leakage into a common rate equivalent, such as gpm covered -- like radiation parameters and all of the other leakage measurement parameters, levels, and so forth, into the leakage parameters, permitted leakage.

That's all I have.

MEMBER ABDEL-KHALIK: I guess I still don't understand the logic of your requirement. What corrective actions do you want him to take for low-level leakage rates for unidentified leaks below the limit?

MR. LI: Yes, that third column is asking them to develop the procedure and evaluate corrective actions. We don't have a set of corrective actions. We don't have, you know, examples so far that we ask -- because they are the first one that has been --

MEMBER ABDEL-KHALIK: But wouldn't the corrective actions depend on the source of the leakage?

MR. LI: Yes.

MEMBER ABDEL-KHALIK: So --

MR. LI: So that before I looked at the bullets, I asked him to first do the monitoring, training, determine the source, and then evaluating, and then develop, analyze and developing the corrective actions. So that's the whole series of things an operator needs to do.

MEMBER ABDEL-KHALIK: But that implies that they have to identify the source of the leak well before the tech spec limit.

MR. LI: That's what we are pushing to. So right now if there is an unidentified leakage of 4.9 gallons per minute, they can just say it's unidentified leakage that's allowed to operate.

MEMBER ARMIJO: But if they get to five, they've got to do -- they've got to shut down or --

MR. LI: They've got to shut down.

MEMBER ARMIJO: They don't want to do that.

MR. LI: They don't want to do that.

MEMBER ARMIJO: They are going to be doing what you want them to do.

MR. LI: Right. It's benefits to them

also when it's one or two gallons per minute, start identifying where the leakage is coming from.

MEMBER ABDEL-KHALIK: But surely tech specs don't go directly from, you know, an unidentified leak of five gallons per minute to a shutdown at 5.01 gpm.

MS. CUBBAGE: Within a pretty short timeframe, they could go --

MEMBER MAYNARD: Well, just about all licensees have an administrative limit that they use before they get there to try to identify what the leakage is, find the source. Once you find the source, as long as it's not a pressure boundary leakage through a crack or something like that, then it becomes identified leakage, which is a higher limit.

But you're not allowed to operate with any leakage of a cracked weld or a pipe or something like that. It can valve leakage, that's okay, or plant leakage, things like that. So typically the process is you see an unidentified leak that's -- whatever your administrative limit is, you work hard trying to find the source of that.

MEMBER ARMIJO: But they are looking for the administrative limit right now, between one and five.

MR. LI: Well, that's not a limit. It's just an alarm to trigger operator to follow all of these actions.

MEMBER ABDEL-KHALIK: I understand.

MR. OESTERLE: Next is SER Section 5.3.1 on reactor vessel materials, and Neil Ray.

MR. RAY: Well, I am Neil Ray with Component Integrity Branch at NRO, and my job is pretty simple. I only talk about reactor vessel and nothing else.

(Laughter.)

And the interesting thing in this case, as you noticed, I was really struggling what to put in here, because GE took almost every of my slides. I mean, I did not prepare it, they did, but that's the thing I was going to talk, but they already talked. So what I am going to talk, I don't know, but let me talk. All right.

CHAIRMAN CORRADINI: We'll come up with something.

MR. RAY: Sure. Please. All right. In terms of reactor vessel materials, as we know, the immediate change as already Jerry talked about it between current BWR and ESBWR -- I am not going to repeat it -- but basically from a regulatory standpoint my job and our job, to make sure that all of the regulatory codes standards, regulatory guidance, in terms of protection of -- and fracture prevention of vessel must be met.

And, in summary, those include 10 CFR 50.55a, 10 CFR 50 Appendix G, 10 CFR 50 Appendix H, and GDC-1430, 31, 32, all those.

And in my review -- actually, in mine and with my colleagues' review, we have gone through this in -- quite in detail, and we asked about at least seven -- between seven to ten RAIs, and all of them are closed, to our satisfaction.

As a matter of fact, what one of you folks asked about the assembly on vessel, we asked the same question during the RAI process, and there was lots of interaction over the phone, and now that pretty much over.

In terms of pressure temperature limits, as we all know that the vessel must be within specific pressure limits, so that there is no such undue fracture. So to follow 10 CFR Part 50 Appendix G criteria, which is basically ASME Section 11 Appendix G. And at this point, since they do not have all of the details in terms of initial activity, or copper, nickel, none of them really available.

The numbers we have gone through or relooked at, basically I call it a kind of conceptual numbers, and those are pretty much good -- as good as gold at this point. It may change; we don't know.

So they provided some conceptual PT limits, and there is a COL action, and they are going to provide the PT limits during the COL stage in formal PTLR, pressure temperature limit report, which will really help them out, because they don't have to put it in tech spec. It will be outside the tech spec. And they can use -- they can refer the PTLR to all other COL applicants.

In terms of operation energy, it is again another number, basically because they don't know anything. However, they have the projected clearance, and using Reg. Guide 1.99 Rev. 2 they projected what would -- how much drop will be in it for a weld. And we all know -- as 10 CFR 50 Appendix G that operation energy got to be at about 50 foot-pound and they said, yes, we'll keep it about 50 foot-pounds, and we are happy.

In terms of high heat also, it will not have any open item except pressure temperature limits. That will be a COL action item.

Now, in terms of reactor vessel integrity, GE-H already provided quite in detail, but, again, in terms of Appendix -- 10 CFR 50 Appendix G, all those things are well met, and we reviewed it and accepted it. The only issue, really COL action item, at this point is the reactor pressure vessel surface capsule program.

And just to refresh all of the people here, that as a matter of fact NRC issued a special inspection procedure for our review and to make sure that reactor pressure vessel surface capsule and capsule holders are in the place as per design. We wanted to make that sure. We don't want to see what happened in the past history from --

MEMBER ABDEL-KHALIK: How accurately can you control the position of those?

MR. RAY: As accurately as we can see in the drawing. Whatever they are going to provide in their surface capsule program, that is the way we are going to inspect it, and that is the way it will be.

MEMBER ARMIJO: That's why you need an as-built dimension, right?

MR. RAY: That is correct. So we are expecting that also, that reactor capsule program will be provided along with the COL folks when they come with the COL application.

I think that's all I have, unless you have any other questions.

CHAIRMAN CORRADINI: So you are not the valve person. I have to go to somebody else for valve --

(Laughter.)

MR. RAY: No, I only talk about vessel.

MEMBER ARMIJO: Is there any issue from the research side about nickel content on these vessel materials? Is there any -- the one percent --

MR. RAY: Yes.

MEMBER ARMIJO: -- seems different than what the GE-H --

MR. RAY: I think what you are referring to, if I understand your question correctly, in the -- pretty much NRC and the Research folks -- Oak Ridge National particularly and Argonne National partly -- they are in the process of revising Reg. Guide 1.99 Rev. 3.

And that answers basically your question. There are questions now -- they have gone all of these years, as you know, if you look back at Reg. Guide 1.99, Rev. 1, Rev. 2, and now Rev. 3 -- it is not done yet. Rev. 1 was strictly based on phosphorous and test reactor. Then, we come with Rev. 2 which is basically based on current reactor database, and we used it.

Now, Rev. 3 is in process, and the question came out -- several questions. Number question is, what is the real, real impact of nickel? What is the real impact of phosphorous sulfur? And the fourth question is: well, we know the vessel when a cumulatively comes to 2.1 to .19 is okay. But if it goes beyond that, how does Rev. 2 treats? And so far the data we have, it does not treat too well. So that's why there is a necessity for Rev. 3, one of the reasons.

We are working at this very moment on Rev. 3. We have -- I don't have any particular target when it will be published or any such thing, but we are working on it.

MR. OESTERLE: According to the public website, Reg. Guide 1.99 is included in Phase 2 of the regulatory guide update program, and they target completion of that at the end of calendar year '08.

MEMBER ARMIJO: Okay. So if there was a, let's say, nickel problem, it may or may not be too late for the guys who

MEMBER SHACK: Some people think nickel is a good thing.

MEMBER ARMIJO: Well, you know, either way -- nickel to be higher or lower, you know, somebody has got to --

MR. RAY: You know, the interesting question you raised, I would add another interesting part to it. Some people ask, "Well, since there is no vessel manufacturing facility in U.S., how about if you make your vessel in Russia?" How are you going to use the Reg. Guide Rev. 2?

You cannot use Reg. Guide Rev. 2 if you manufacture vessel in Russia, because their vessel content is high phosphorous. Reg. Guide Rev. 2 does not address high phosphorous. You cannot do it. You have to develop a completely different methodology.

Now, in terms of what you are saying, we already know -- probably "know" is not a right word -- we probably can speculate what will happen when Rev. 3 really comes out to -- for public or for utilities' usage -- GE-Westinghouse folks. Our initial reaction is the -- it doesn't sound good.

The delta P -- delta will be -- in most cases will be higher than what we are used to today. That is our initial reaction.

MEMBER ARMIJO: Okay. Sorry I asked.

(Laughter.)

MEMBER SHACK: You don't want to unleash a vessel guy.

(Laughter.)

MR. OESTERLE: And on that positive note, we will move on to Section 5.4.6, isolation condenser system.

MR. THOMAS: Okay. We reviewed the system according to standard review plan 5.4, 6, and 6.3. And this ICS system is part of the emergency core cooling system, and GE takes credit for the liquid in the condensate line. So that's why it is different than the current operating plants in our -- Oyster Creek, Nine Mile -- that's when they all got IC, but this is the big difference in this.

And we had a concern about this -- during this, and we had questions on this issue. And that since IC comes on material than the DPV, and by the time the faster DPV opens, all that currently will be already gone from there. And physically also, there is some distance all of these. We are convinced that there is no issue now, so initially we -- you had questions about this.

CHAIRMAN CORRADINI: I don't think I appreciate what you just said. Could you --

MR. THOMAS: You know, in the stop tube or stop line, they are sharing this DPV line and the --

CHAIRMAN CORRADINI: Right.

MR. THOMAS: -- IC line.

CHAIRMAN CORRADINI: Right.

MR. THOMAS: And basically, they are in the same line.

CHAIRMAN CORRADINI: Right.

MR. THOMAS: So we were very concerned about the -- you know, how this will interact during a LOCA. Okay? And if you talk about the LOCA scenario, IC comes on -- because it's high reactor pressure, around 1,080 psig. So by the time DPV opens, IC will be already doing the -- most of the job in the beginning of the accident. Okay? So that issue is complete now, resolved now.

CHAIRMAN CORRADINI: Primarily because of timing, because of the --

MR. THOMAS: Right, right.

CHAIRMAN CORRADINI: -- phasing.

MR. THOMAS: Right, right.

CHAIRMAN CORRADINI: Right. Thank you. Okay.

MR. THOMAS: Right, right, right.

CHAIRMAN CORRADINI: I got it.

MR. THOMAS: And we have some concerns about some operational issues also, because from the experience there were so many licensee event reports. There were about 150 of them. We had research done by our contractors during the last time that we did the ESBWR, by Oak Ridge, and they said there were 150 events during a 10-year period.

And we were concerned about that, and we had questions to GE. And GE told us that they made improvements from the old design. They changed the material of the piping, they changed the material of the tubing, and they also did some --

MEMBER SHACK: Yes. I mean, what was the nature of these events? It was problems with corrosion and cracking?

MR. THOMAS: There were so many problems, but I am -- according to GE, they went through all of that, and then they gave it improvements. Okay? So I was talking about three improvements.

One is that they changed the piping material, because they had this IGSCC, you know, the inter-granular stress corrosion cracking. So they changed the material of the -- of the piping in the ESBWR. Also, they changed the material of the tubing in the IC. And also, they did the -- the swapping of the condensate line. Okay? So they made some improvements from the current problems in the old plants like Oyster Creek and Nine Mile and North Anna, and all of that.

So we were learning from that experience, so that, you know, those problems may not happen in the ESBWR. So --

MEMBER ABDEL-KHALIK: Back to the issue of the interaction between the ICS and the DPV --

MR. THOMAS: Right.

MEMBER ABDEL-KHALIK: -- is it still not a problem if you have inadvertent failure of the DPV?

MR. THOMAS: It is a very -- it can happen, but I -- you know --

MEMBER ABDEL-KHALIK: I mean, one of the valves can fail open.

MR. THOMAS: That's a very small flaw, and it is still bounded by the main steam line break at --

MEMBER ABDEL-KHALIK: And how would that affect the operation of the ICS, if they are connected to the same line?

MR. THOMAS: ICS will still function, but the steam flow will be less here.

MEMBER SHACK: But isn't the ultimate safety system in the LOCA the gravity system anyway, so you'd be -- you'd blow the DPV --

MR. THOMAS: The LOCA is -- there are four of them, and you've got only three of them for the safe shutdown or LOCA or any event.

CHAIRMAN CORRADINI: Right. But I think what Dr. Shack was saying, though, is if you get in a situation where you have the DPV open, you want to go to low pressure, so you're essentially going the direction you want to engage your other --

MR. THOMAS: Right.

CHAIRMAN CORRADINI: -- your other system.

MR. THOMAS: Right. Well, you've got a number of open items in this section, but mostly they are related to material issues and ISI and all of that, so --

MEMBER SHACK: Well, it's nice to know that you think the system is going to work.

(Laughter.)

MR. DAVIS: I think earlier somebody asked if they were going to revisit the isolation condenser, and that is part of Chapter 6.

CHAIRMAN CORRADINI: So since we're on what's where, where is the MSIV? It's not in Chapter 5. Is it going to be in Chapter 6, to describe how its operation is? Did I miss it?

MS. CUBBAGE: This is Amy Cubbage. It's Chapter 3.

CHAIRMAN CORRADINI: Ah. Okay, fine. I should have known that.

MS. CUBBAGE: But as far as when we would talk more about the isolation condenser, with respect to the water volume available in a LOCA, that would be Chapter 6. But that's -- there won't be a lot of discussion about that. But as far as --

CHAIRMAN CORRADINI: But I've got a funny feeling that other members of our Committee who aren't here, but will be excited about the analysis of how it actually takes away the heat during some sort of --

MS. CUBBAGE: Right. And that's not part of Chapter 6, because the assumption for Chapter 6 is just that water volume that's credited in the LOCA analysis, you might be interested in Chapter 21 where we might speak about the test programs and the validation of the TRAC-G code.

CHAIRMAN CORRADINI: That's fine. This is how you make it. This is how it necessarily operates.

MS. CUBBAGE: Right. If you're concerned about how the system works, this is the chapter to discuss it. If you're interested in the test program, that's Chapter 21.

CHAIRMAN CORRADINI: Okay.

MR. THOMAS: There is no RHR system in ESBWR. It is all done by RWCU. Pressure down-cooling is done by RWCU.

And we had some questions -- we are still waiting from GE-H. You had a question about the thermal mixing, and we also got questions about the heat removal capacity of the RWCU, so it is still open. We are waiting from -- from GE about this.

And since the system is designed to the full reactor pressure, the whole system is designed for 1,250 psig. So there is now concerns about the intersystem LOCA like we had before RHR system. We had a low pressure system and high pressure, and we had a lot of problems before.

And we are going to high point vents -- there is something continuous going on in the ESBWR that is vent line going to the main steam line. So it is continuously vented. And there is only a vent line for the shutdown conditions, so during the shutdown you can open the vents that go to the drain system. But those were -- but there is another line going to the main steam line, and all of the gases -- non-condensable gases are all removed through the off-gas system.

So this system is not part of the emergency core cooling system. So venting in ESBWR is all done by the safety relief valves, so this is not a part of the emergency core cooling system. So there are no open items in this section, so it is complete and --

MR. OESTERLE: Are there any questions on that last section?

MEMBER SHACK: I just -- Reg. Guide 1.56 --

MR. THOMAS: Oh. I want to include one more thing about the cleanup system. You know, there is Reg. Guide 1.56. I'm referring to Reg. Guide 1.56 and the EPRI report.

MEMBER SHACK: Why don't you trash 1.56. I mean, it's 1975 vintage. It assumes -- it really doesn't provide an acceptable water chemistry specification for a BWR any longer. Any reference to it should be deleted.

MR. OESTERLE: One of the things that happened with the submittal of the DCD, at the time it was submitted, it was prior to the SRP update program that we engaged in, and the DCD referenced Reg. Guide. 1.56. And when this section was reviewed, the SRP that existed at the time the staff reviewed that section still referenced Reg. Guide 1.56.

We understand that probably COL applicants will go with the EPRI reports, but that is something that we're taking a look at, as we have gone to a full update of all of the SRPs.

In conclusion, we can see that considerable progress has been made towards resolving a number of the staff's requests for additional information. The staff continues to engage GE in discussions to resolve open items and additional RAIs. And although the staff believes that GE-H is making progress towards an acceptable design, there are still a number of open items that remain to be resolved. And as a result, the staff at this point is unable to finalize all of the issues on the reactor coolant system and connected systems.

The staff is still looking at -- and it has begun reviewing -- Rev. 4 of the DCD. And we have seen so far, it appears that some of the open items will be resolved by Rev. 4 of the DCD, but we are still progressing with that review. In summary, the staff looks forward to presenting the resolutions to these open items as part of its presentation to the ACRS of its final safety evaluation report on the ESBWR design certification in the future.

MEMBER SHACK: Are you still sending out RAIs on this?

MR. OESTERLE: Yes, sir, we are.

CHAIRMAN CORRADINI: On all chapters, as you get updates from the applicant, you are sending out -- I mean, for example, on Chapter 5, still others are going out.

MR. OESTERLE: Yes, that's correct. And as we review Rev. 4, some of the open items that we currently have identified may be resolved.

CHAIRMAN CORRADINI: Or closed.

MR. OESTERLE: Right. However, some information may result in new RAIs.

CHAIRMAN CORRADINI: Okay.

MR. OESTERLE: Other questions from the Subcommittee?

(No response.)

CHAIRMAN CORRADINI: So I guess I -- well, I wanted to thank all of you, but I have a question for maybe not you, but I will turn to Amy and ask. So I -- a couple of the Committee members that aren't here that were at last one -- and I'm thinking of Chapter 8, for example -- brought up the concern of system interactions. That is, we might be looking at part of the system here and it looks fine.

The open items, as you have identified them, and what you're seeking as additional information, seem reasonable. But down the road, something may happen from -- something may come up in our minds about an interaction between one part of the system and another. I don't think isolation condenser performance is a good example, but that's the only one I can come up with as a bad example.

And as we look at maybe accident progression, or something else that refers us back, how are we to handle -- how do you want us to handle this relative to -- is there going to be a final kind of roll up of system interaction discussions between the various systems that we -- the Committee can talk about with the staff? How do you want to do that?

MS. CUBBAGE: Yes, this is the SER with open items phase, of course. We'll be coming back with the final SER.

At this stage, we're interested in hearing any comments that you hear based on our presentation and the material you have available. So we're looking for that early feedback.

So if you do come up with a question later that relates back to this, we are just looking to hear that from you as soon as you identify it.

CHAIRMAN CORRADINI: It may not be an open item, but it may be a comment that concerns us about something. The one that pops in my head is -- is John Stetkar's discussion where he was more worried about HVAC, which is not what we're reviewing yet, but how it may feed back and impact electrical systems if it happens to be an overheating of a key system that we are going to need for DC power.

MS. CUBBAGE: Right.

CHAIRMAN CORRADINI: So those sorts of comments we can -- we will probably include, but they are not necessarily open items as much as current worries.

MS. CUBBAGE: I understand what you're saying now. So for this meeting, if you were to identify an issue that didn't specifically relate to the chapters you've seen, we would like you to identify those to us in your interim letter, so that we could be prepared to address those --

CHAIRMAN CORRADINI: Okay.

MS. CUBBAGE: -- when we come back.

CHAIRMAN CORRADINI: Okay. That at least satisfies some of my concerns. I'm trying to figure out how to handle some of the things that may pop up.

Other questions? Can we just go around

and just see if anybody has additional questions? And since we're not only talking about these three, but previous three, just things that I can write down. All right?

MEMBER ABDEL-KHALIK: Well, with regard to Chapters 11 and 12, I really would like to see a better justification for the source term that was used.

MS. CUBBAGE: Right. And as a matter of fact, before -- maybe before we get into the full conclusion here, we do have two staff members that have returned, hopefully with a little additional information in that area.

Jai, would you like to step up? Well, I don't know. Do you have questions of this group

CHAIRMAN CORRADINI: I don't think so. They're not going to go far.

MS. CUBBAGE: Okay.

(Laughter.)

They'll be here. Okay.

CHAIRMAN CORRADINI: So this addresses Said's question about the source term? Okay.

MEMBER ABDEL-KHALIK: Would you like for me to pose the question one more time, or --

MR. LEE: This is Jai Lee, responsible for source term and this chapter. We understood your question, and we prepared a draft response for you.

MEMBER ABDEL-KHALIK: Thank you.

MR. LEE: And we'd like to have you --

CHAIRMAN CORRADINI: But ours wasn't an RAI, though.

(Laughter.)

MR. LEE: Maybe you can quickly read this over, and then we are here to answer any more -- further questions you may have.

CHAIRMAN CORRADINI: Okay. Go ahead.

MS. CUBBAGE: Why don't you summarize it.

MR. LEE: I'm just going to read what we have prepared. And, you know, actually, when we said one percent fuel detect, that means one percent of the fuel rods in the reactor core will experience some degree of fuel cladding defects. So this is not really any numerical value, but actually the 100,000 microcurie per second release is really covered in the source term.

We further stated here that it is recognized that one percent fuel rod defect does not represent one percent of a core, or gas gap inventory in the radioactivity levels. So it has nothing to do with the fission product inventory in the core. We simply meant one percent of fuel in the reactor core will experience some kind of a fuel defect, such as maybe pinhole type leakage.

So in the context of a presentation this morning, the slide on Chapter 11 perhaps should not have included this one percent number in it. It is a more conceptual number, to give you some background information.

So the basis for accepting 100,000 microcuries per second release proposed by GE-H is really based on our NUREG-0016. The default noble gas release rate is assumed to be 55,000 microcuries per second at the 30-minutes decay. And it is normalized to the 3,400 megawatt thermal in the NUREG-0016.

This is based on actual measurement of operating 13 BWR -- excuse me, 12 BWRs. And we -- you can find that reference in NUREG-0016, page number --

MR. KRESS: Is this a mean number, or is it a bounding number? Or are these 13 BWRs?

MR. LEE: This is 13 BWR average number.

MR. KRESS: Average number.

MR. LEE: Right. It's not bounding.

MR. KRESS: Now, is there a very big range of that?

MR. LEE: It ranged from anywhere -- I

believe starting from 1,000 to the -- actually, it approaches to the 100,000 microcurie per second release rate.

MR. KRESS: Now, that's a function of the -- what gets out of the fuel, but it is also a function of how much mass of water is in the RCS, and how fast the stuff is removed and the various removal --

MR. LEE: Right. It's --

MR. KRESS: -- like the charcoal beds or the -- well, this is noble gases, maybe through the off-gas system.

MR. LEE: Right. Hold-up system. This is really 30-minutes decay through the hold-up --

MR. KRESS: That's what a 30-minute decay equates to.

MR. LEE: Yes.

MR. KRESS: Now, those things are different for these different BWRs?

MR. LEE: I believe GE -- maybe GE can answer, but its hold-up system is essentially the same as could an operating BWR to the -- this ESBWR. I don't think they changed, basically, any -- any basic design.

MR. McCULLOUGH: I can speak to that.

MR. LEE: Sure.

MR. McCULLOUGH: Dale McCullough. The hold-up system in the ESBWR is actually bigger than the current operating BWRs.

MR. KRESS: So it would be conservative to assume this number.

MR. McCULLOUGH: Yes.

MR. KRESS: Okay.

MR. McCULLOUGH: Yes.

MR. KRESS: Okay. Now, the other question about that is: was the scale-up done strictly by reactor power?

MR. LEE: Yes, that's what we did.

MR. KRESS: Is there a technical basis for that, or an experience base?

MR. DEHMEL: The technical basis is basically reflected in the BWR GALE Code as well as the ANSI Standard 18.1-1999. That essentially is the --

MR. KRESS: Is that GALE Code -- I'm not familiar with it. Has it been reviewed and approved by you guys?

MR. DEHMEL: Yes. It has -- the issue has been around by the NRC for a while now. It has been slated for revision and update.

MR. KRESS: So the code will tell you that the release rate -- so basically related to the release rate. And it is proportional to the power level, is that what the GALE Code would tell you?

MR. LEE: Yes. Power level will be directly proportional to the fission product inventory in the core.

MR. KRESS: Sure.

MR. LEE: So the bigger the core, why, we expect to release more. So we assume here it is directly proportional to the reactor power.

MR. KRESS: Well, I know it's an assumption, but is this a -- is there a -- has it been validated with some sort of an experience base?

MR. DEHMEL: We are in the process of kind of getting to the underlying line of questioning. We are in the process of embarking on the revision of the BWR GALE Code and PWR GALE Code just for that purpose, because we feel that all of the operating practices that have been folded into this BWR GALE Code and PWR Gale Code and NUREG-0016 and 007 reflects the operating history of the earlier fleet of powerplants.

And we are going to revise this, including the supporting reg. guide and Reg. Guide 1.2. So there is an effort afoot to actually undertake this.

And, essentially, if you look at the basis section of both NUREGs, which is based on operating practices as well as studies that GE and Westinghouse did with specific plans, from which these values came about -- for example, the ones that just -- Jai Lee cited, we are going to look at this and try to figure out how to update this to essentially reflect, you know, the current fuel performance, current reactor design, improvement in process treatments of both liquid and gaseous effluents, and so on.

At this point, we are kind of stuck with the regulatory tools that we have.

MR. KRESS: You have to go with what you've got. Yes, I understand that.

CHAIRMAN CORRADINI: Do you have a further question?

MEMBER ARMIJO: Just a question on -- as long as we're talking about fuel, when is -- when are we going to review the -- as a Subcommittee the core fuel chapters?

MS. CUBBAGE: Right. The fuel is Chapter 4, and we are planning for a January Subcommittee meeting on fuel.

MEMBER ARMIJO: Is that news to you, Mike?

CHAIRMAN CORRADINI: No.

(Laughter.)

It didn't really sound like a question.

MS. CUBBAGE: Although I will add that this is not a big stretch from the operating fleet, they are going to be using very similar fuel. It is just shorter in length than the operating fleet.

MEMBER ARMIJO: Well, as long as we're on it, I kind of scanned that chapter, and I noticed that there was no mention about a built-in PCI resistance in the description. I wondered if GE-H had changed their concept on that.

MS. CUBBAGE: I don't know if they have anyone here to speak to that. I know there are separate topical reports that supplement the DCD that may get into a higher level of detail than what you are seeing.

MEMBER ARMIJO: Well, it is what is in the text of the report.

MR. UPTON: Sam, this is Hugh. We have not changed the fuel design. We are still planning on using barrier fuel, so it will be PCI-resistant.

MEMBER ARMIJO: Or some alternative.

MR. UPTON: Or some alternative, yes.

MEMBER ARMIJO: Okay.

CHAIRMAN CORRADINI: Any other comments?

Bill?

MR. LEE: Okay. Like we said, we further scaled up this -- it happened to be 73,000 microcuries per second release rate at the 30-minutes decay for the ESBWR design at the 4,500 megawatt thermal, and which is bounded by 100,000 microcuries per second release rate that GE proposed.

MR. KRESS: Now, this scale-up, that is your ratio of power --

MR. LEE: Ratio of powers to power.

MR. KRESS: -- some sort of mass to water.

MR. LEE: I did not use mass. Even though GE has -- I mean, ESBWR has larger inventory of water, and also it has higher steam flow rate, as well as higher cleanup rate, but the ratio is I used just thermal power.

MR. KRESS: Just the thermal power.

MR. LEE: Right.

CHAIRMAN CORRADINI: That's that essentially -- then, it's some source and some whole --

MR. LEE: Just inventory.

CHAIRMAN CORRADINI: -- that's why you're -- it's just an inventory issue. Yes?

MR. KRESS: Yes. Well, I looked at that

in the -- I guess it's in the design and control document somewhere. It bothered me the scale-up had decay constant times the total amount of that radionuclide as a removal, which is all right. But it also had a removal rate which was proportional to the total amount in the water, which seemed a little strange to me. I still have some worries about that.

But you didn't use that. You just used the ratio of power. So that does away with my issue there, but I -- I think that equation needs to be looked at. It doesn't seem reasonable to me that the removal rate is proportional to the total amount in the water. It ought to be proportional to the concentration.

I think we need to look at that equation, but it doesn't impact here because --

CHAIRMAN CORRADINI: Okay.

MR. KRESS: -- he finessed it.

MR. LEE: Okay. So that was our basis. Then, GE-H, they stated that the 100,000 microcuries per second release rate at the 30-minutes decay is based on their particular topical report. And so maybe GE-H can address this topical report -- your basis for 100,000 microcuries per second release rate.

MR. KIRSTEIN: This is Eric Kirstein with

GE-H. The 100,000 microcuries per second was based on the NEDO reporting in question as it was generated in 1971. Data were taken for release rates, noble gas release rates, from I think -- I believe 1968 through 1971. Values were generated on the order of .1 curies per second, 100,000 microcuries per second.

This value was deemed to be a good representative value for noble gas release rates, given the characteristics of the fuel at the time. Stainless steel clad fuel met multiple fuel failures.

MR. KRESS: Do you think it is probably overly conservative?

MR. KIRSTEIN: I believe so, yes. Yes.

CHAIRMAN CORRADINI: Are you all -- do you want to stop here?

MR. KRESS: Were the other radionuclides, like iodine and so forth, based on the same kind of experience?

MEMBER ABDEL-KHALIK: Mr. Chairman, I am satisfied.

CHAIRMAN CORRADINI: thank you.

MR. KRESS: Yes, I'm okay with this, although I think we ought to look at that equation.

CHAIRMAN CORRADINI: Other questions from the Committee? Sam?

MEMBER ARMIJO: I have a comment.

CHAIRMAN CORRADINI: Go ahead.

MEMBER ARMIJO: And it's not related to fuel or --

CHAIRMAN CORRADINI: That's fine.

MEMBER ARMIJO: It is related to the fabrication requirements. I believe both GE-H and the staff should relook at this issue of post-weld grinding and tolerance of post-weld grinding.

For an industry that spends hundreds of millions of dollars to billions of dollars on responding to cracking events, we should have learned enough by now that we can set some very high standards on the quality of initial welds and relook at the issue of radiography to get a nice picture, but in allowing grinding, which will just give us a nightmare as far as cracking downstream. So I would encourage the staff to rethink that.

MR. LEE: Yes, we will.

CHAIRMAN CORRADINI: So can I -- since I have the -- I have a funny feeling I'm going to be writing this up -- can I ask you to encapsulate that in a few --

MEMBER ARMIJO: Sure.

CHAIRMAN CORRADINI: -- choice words?

MEMBER ARMIJO: Yes. I'll write something up for you.

CHAIRMAN CORRADINI: Thanks. I think I understand your point, but I want to get --

MEMBER ARMIJO: Yes.

CHAIRMAN CORRADINI: -- some more basis.

MS. CUBBAGE: Mike, before we continue on, we did have one more followup item from this morning.

CHAIRMAN CORRADINI: Oh, I'm sorry.

MS. CUBBAGE: And that was on the AAOs in the context of Chapter 11.

CHAIRMAN CORRADINI: Okay.

MS. CUBBAGE: And I believe the staff and GE both have something to say on that. I'll start with Jean-Claude.

MR. DEHMEL: Yes. Jean-Claude Dehmel. This is a followup to some of the questions during our presentation on Chapters 11.2 and 11.3 regarding a definition of the anticipated operational occurrences.

In this handout, I cite the source where there is a definition of anticipated operational occurrences, and basically it simply says the term means unplanned releases of radioactive materials from miscellaneous actions such as equipment failures and operator errors that are not of consequence to be considered an accident. So just to reinforce the point what I said this morning. And also, that the NUREG does not provide a list or a catalog of what are typical AAOs.

MEMBER ABDEL-KHALIK: But do we have a database from operational history?

MR. DEHMEL: In the back of NUREG-0016 is a very small history of AAOs, and basically, from what I recall, nearly 60 percent are due to operator errors, 26 percent are due to equipment failures, and then there is a small description of other types of nondescript type of events and mishaps, and that's it. There is no real database.

I mean, there is one, but it would involve scouring all of the inspection reports from each plant, each docket, and develop a database that would essentially track there. But there is no current -- more current database for this.

MEMBER ABDEL-KHALIK: That's all I have.

MS. CUBBAGE: GE, were you going to speak on this?

MR. TUCKER: Does that description close any remaining questions from this morning, or is additional discussion needed?

CHAIRMAN CORRADINI: I think I see -- at

least at the current level of satisfaction from that side of the table, yes.

MR. TUCKER: Thank you.

CHAIRMAN CORRADINI: Let me turn over here. Otto, did you have comments about any of the chapters?

MEMBER MAYNARD: I did have one question I should have asked GE when they were up here on the vessel. It's a large vessel, very long vessel. Are all of the internals in the vessel are going to be manufactured in the same location?

Let me tell you where I'm going with this. It's such a long vessel, parts being assembled and built in different places. Any temperature gradients -- when you get to assembling this thing, it may not fit even though they were built to tolerances at the location. What controls, or what are you doing to make sure that it really does fit when put together in the same location?

MR. DEAVER: Okay. This is Jerry Deaver. Basically, the vessel scope of supply will be the vessel and the shroud support structure. All of the internals, you know, because of the heavy nature of them, will be site-assembled. That's part of the details that we will be looking at. You know, is the assembly tolerances and -- of the structures as we go through the detailed design of all of the components, which we are activity working on right now. So --

MEMBER ARMIJO: Would that include the shrouds and chimneys, and everything else --

MR. DEAVER: Oh, yes.

MEMBER ARMIJO: -- would be separate?

MR. DEAVER: Right. You know, some of them are bolted. We just have to make sure that -- you know, a lot of the fit-up will be basically establishing an alignment, and then locking it in place. You know, bolting will have some amount of tolerance for fit-up and then -- but once the permanent alignment is established, then we will be able to lock it in place, you know, by match-drilling holes and so forth.

So, I mean, I don't see any big issues, you know. It is a taller structure, but it's something we -- we know we're going to have to pay attention to as far as the details on alignment.

MEMBER MAYNARD: Just from some personal experience, I was working at Boeing when the 747 came out. Parts were built in Wichita, some were built in Seattle. When they put them together they didn't fit. They had not had trouble with other airplanes that were smaller, but it was such a larger airplane that those differences weren't showing up, and it created some problems until they resolved it.

MR. DEAVER: Yes. Typically, in our procedures and that, we have talked in terms -- when we take as-built dimensions and stuff, they have to be at nominal temperatures and stuff or corrected. So --

MEMBER MAYNARD: Don't take it too lightly. It's a big piece of equipment to get all together.

MR. DEAVER: We've got a lot of big parts.

MEMBER MAYNARD: I don't have anything else, Mike.

CHAIRMAN CORRADINI: Tom?

MR. KRESS: I have said all I needed to say.

CHAIRMAN CORRADINI: For today.

MR. KRESS: Yes.

CHAIRMAN CORRADINI: Okay. Do you have any last comments, Amy?

MS. CUBBAGE: Well, no. I guess we -- both the staff and GE-Hitachi would be interested in knowing what the scope of the presentation the Committee is interested in on November 2nd for the full Committee.

CHAIRMAN CORRADINI: I guess I have given that some thought. I'd like to hear from the other members, but my feeling is that with the whole Committee present, thinking just based on the comments I've gotten electronically by some of the other Committee members, I think others will probably be coming back to ask additional things about Chapter 2 and siting. Probably something about Chapter 5 materials.

I mentioned already system interactions from a couple of the members, but I don't think those are chapter-specific. And other than that, I don't see any major issues from Chapters 11 and 12. We may come back and talk about source term again, but I don't think that is 11 or 12 specifically. That is more generic.

But I was going to say that I don't see anything from --

MEMBER SHACK: Well, I mean, that source term is really only for the --

CHAIRMAN CORRADINI: From coolant.

MEMBER SHACK: Yes.

CHAIRMAN CORRADINI: Right. But I guess what I'm getting at is I don't see anything from Chapters 8 and 17 and 11 and 12 that are specific that would demand additional people. But I think -- of the two, I think 2 and 5, you are probably going to get additional questions, if that's what you're --

MS. CUBBAGE: We are planning to bring --

CHAIRMAN CORRADINI: -- planning on coming with.

MS. CUBBAGE: Right. We are planning to bring technical staff for all of the chapters that we have discussed previously. And we just wanted to tailor our presentation accordingly.

CHAIRMAN CORRADINI: I guess we can talk about it afterwards, but -- let's talk about it afterwards.

MS. CUBBAGE: Sure.

MEMBER ARMIJO: Mike, there's one thing that maybe Bill and I were both interested in is to get the material specs from the project data book as opposed to the design control document, whatever level of detail exists.

CHAIRMAN CORRADINI: Do you want that before next week? There's no issue to that.

MEMBER ARMIJO: No. No. Just so long as we get it.

MEMBER SHACK: I mean, I can sort of understand their desire not to put some things in concrete, even though they are --

MEMBER ARMIJO: Yes, I understand.

CHAIRMAN CORRADINI: Any other things?

(No response.)

Well, let me thank GE-H for their time and efforts. It was very helpful. And the staff -- thank you, Amy, and all the staff.

MS. CUBBAGE: Thank you.

CHAIRMAN CORRADINI: All right. And I guess we're adjourned.

(Whereupon, at 3:31 p.m., the proceedings in the foregoing matter were adjourned.)

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.

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Presentation to the ACRS Subcommittee

ESBWR Design Certification Review

Chapter 5
Reactor Coolant System and Connected Systems

October 25, 2007

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 5

Purpose

- Brief the Subcommittee on the staff's review of Revision 3 of the ESBWR design certification application, Chapter 5, "Reactor Coolant System and Connected Systems"

- Answer the Committee's questions

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 5

<u>Review Team for Chapter 5:</u>

- Lead PM (DNRL)
  - Eric Oesterle, Senior Project Manager
- Lead Technical Reviewers
  - Cheng Ih Wu (5.2.1)
  - George Thomas (5.2.2, 5.4.6, 5.4.7, 5.4.8, 5.4.12)
  - Robert Davis (5.2.3, 5.2.4)
  - Chang Li (5.2.5)
  - Neil Ray (5.3.1, 5.3.2, 5.3.3)
- Secondary Reviewers
  - Yamir Diaz-Castillo
  - Lambros Lois
  - John Fair
  - John Huang

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 5

- Applicable Regulations
- RAI Status Summary
- SER Technical Topics
- Open Items
- COL Action Items
- Discussion / Committee questions

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 5

Summary of Regulations and other Review Guidance

- 10 CFR 52 Subpart B – Standard Design Certifications
- 10 CFR 50 applicable sections and appendices
  - Appendix A, General Design Criteria
  - Appendix B, Quality Assurance Criteria for Nuclear Power Plants…
  - Appendix G, Fracture Toughness Requirements
  - Appendix H, Reactor Vessel Material Surveillance Program Rqmts
- GDCs: 1, 2, 4, 5, 14, 15, 17, 19, 30, 31, 32, 33, 34, 36, 55, 60, 61
- Regulatory Guides: 1.26, 1.29, 1.31, 1.34, 1.36, 1.37, 1.43, 1.44, 1.45, 1.50, 1.56, 1.65, 1.71, 1.84, 1.99, 1.139, 1.147, 1.190, 1.192
- SRPs: 5.2.1.1, 5.2.1.2, 5.2.2, 5.2.3, 5.2.4, 5.2.5, 5.3.1, 5.3.2, 5.3.3, 5.4.7, 5.4.8, 5.4.12, 5.4.13 (5.4.6 and 6.3)
- Other guidance (generic communications, NUREGs, and SECY's)

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 5

RAI Status Summary

- Original RAIs: 138
- RAI's resolved: 118
- Open Items: 20
- Significant Open Items to be discussed later in the presentation

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 5

5.2 Integrity of Reactor Coolant Pressure Boundary
- 5.2.1 – Compliance with Code and Code Cases
- 5.2.2 – Overpressure Protection
- 5.2.3 – Reactor Coolant Pressure Boundary (RCPB) Materials
- 5.2.4 – RCS Pressure Boundary Inservice Inspection and Testing
- 5.2.5 – Reactor Coolant Pressure Boundary Leakage

5.3 Reactor Vessel
- 5.3.1 – Reactor Vessel Materials
- 5.3.2 – Pressure-Temperature Limits
- 5.3.3 – Reactor Vessel Integrity

5.4 Component and Subsystem Design
- 5.4.6 – Isolation Condenser System
- 5.4.7 – Residual Heat Removal System
- 5.4.8 – Reactor Water Cleanup/Shutdown Cooling System
- 5.4.9 – Main Steamlines and Feedwater Piping
- 5.4.12 – Reactor Coolant High Point Vents

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 5 – SER Section 5.2.1

Compliance with Codes and Standards

- The staff has concluded that the design of all ESBWR ASME Code, Class 1, 2, and 3 components and their supports will conform to the appropriate ASME Code Edition and Addenda, ASME alternate Code Cases and Commission's regulations in accordance with 10CFR50.55a

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 5 – SER Section 5.2.2

Overpressure Protection

- 10 SRVs and 8 SVs provided for overpressure protection in the ESBWR design
- Staff requested GEH to address several operational issues related to SRV performance
- The SRVs and SVs will comply with ASME Section III and XI
- Peak calculated pressure is 1263 psig which is below 1375 design pressure of RCS
- COL Item: overpressure protection analysis for the initial core

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 5 – SER Section 5.2.3

RCPB Materials – Ferritic Materials

- With the exception of satisfactory resolution of the Open Items, RCPB materials were found to comply with the requirements of ASME Code, Section III
- All ferritic materials used in the RCPB conform to ASME Code, Section III
- Ferritic materials (i.e. piping, components, bolting) meet fracture toughness requirements of ASME Code, Section III

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 5 – SER Section 5.2.3

RCPB Materials – Austenitic Stainless Steel Materials

- All austenitic stainless steels are supplied in the solution heat treated condition
- All austenitic stainless steels to be welded are low carbon (< 0.02%C)
- Materials and processing conform with the guidance in NUREG-0313, Rev. 2 and RG 1.44
- Cleaning and cleanliness controls conform with the guidance of RG 1.37

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 5 – SER Section 5.2.3

Reactor Coolant Pressure Boundary Materials

- RCPB materials are compatible with reactor coolant water chemistry

Significant Open Items

- Material specification for Class 1 feedwater valves
- Use of ASTM A800 to calculate ferrite content of cast austenitic stainless steels in lieu of Hull's Equivalent Factors
- Filler metal specifications for welding low alloy steel feedwater components
- Justification for attaching ASTM A709 HPS70 to the containment liner
- No COL Action Items for Section 5.2.3

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 5 – SER Section 5.2.4

Preservice and Inservice Inspection of RCPB

- With the exception of Open Items, previously identified PSI and ISI of RCPB was found to comply with the requirements of 10 CFR 50.55a and ASME Code, Section XI

- Development of the preservice inspection (PSI) and inservice inspection (ISI) programs is the responsibility of the COL Holder

- PSI and ISI ultrasonic examinations will be performed in accordance with the ASME Code, Section XI, including conditions specified in 10 CFR 50.55a

- All items within the Class 1 boundary are designed to provide access to perform PSI and ISI examinations required by ASME Code Section XI, IWB-2500

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ESBWR Design Certification Review
Chapter 5 – SER Section 5.2.4

Accessibility to Perform Ultrasonic Examinations

- For piping, pumps, valves, and supports, welds are designed to permit ultrasonic examination from at least one side
- Where component geometries permit, access from both sides is provided
- GEH has indicated that Radiography may be used for PSI and ISI
- The staff is concerned with this approach because current operating plants often seek relief from performing radiography

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 5 – SER Section 5.2.4

Significant Open Items

- ISI of austenitic and dissimilar metal welds and proposed use of radiography
- Examination of reactor pressure vessel to nozzle welds using NRC endorsed Code Case N-613-1
- Modify DCD to include a COL Action Item to fully describe the PSI and ISI programs and augmented inspection program

COL Action Items

- COL Holder is responsible for the development and implementation of the PSI and ISI program plans that are based on ASME Code Section XI

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 5 – SER Section 5.2.5

RCPB Leak Detection

- Open Item 5.2-1: detection alarm setpoint of 5 gpm for unidentified leakage should be reduced for consistency with RG 1.45, Rev. 2
- Open Item 5.2-2: alarm setpoint should be reduced to alert operators to implement procedures to manage the leakage prior to reaching TS limit of 5 gpm
- COL Holder Item: develop procedures for monitoring, trending, determining leakage source, and evaluating corrective actions for low level leakage rates
- COL Holder Item: develop procedures to convert different sources of leakage into a common rate equivalent (gpm)

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 5 – SER Section 5.3.1

Reactor Vessel Materials

- Reactor Vessel Materials are acceptable
  - Complies with ASME Section III
  - Complies with 10 CFR Part 50, Appendix G
  - Complies with GDC 1 and GDC 30
  - Complies with 10 CFR 50.55(a)(1)
- Per DCD, RV maximum material limits
  - Base Material: 0.05% Cu, 0.006% P, 1% Ni (forging)
  - Weld materials: 0.05% Cu, 0.008% P, 1% Ni, 0.008% vanadium
- No Open Items

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 5 – SER Section 5.3.2

Pressure-Temperature Limits

- Pressure-Temperature Limits will be developed in accordance with 10 CFR Part 50, Appendix G and Regulatory Guide 1.99
- COL applicant will provide the pressure-temperature limits including the methodology

Upper-Shelf Energy

- Predicted drop in upper shelf energy for welds, 6.6 ft-lb
- Predicted drop in upper shelf energy for beltline forging, 15 ft-lb
- End-of-life upper shelf energy, > 50 ft-lb

No Open Items for Section 5.3.2

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 5 – SER Section 5.3.3

Reactor Vessel Integrity

- Design, materials, fabrication, inspection, and quality assurance requirements will conform to applicable NRC regulations and Regulatory Guides, as well as requirements of ASME Code Section III
- Fracture toughness requirements of 10 CFR 50, Appendix G
- Reactor vessel material surveillance program will capture the change in fracture toughness properties of reactor vessel beltline material properties (COL Action Item)
- No Open Items for Section 5.3.3

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 5 – SER Section 5.4.6

Isolation Condenser System

- Part of emergency core cooling
- Credited 3 out of 4 for LOCA analysis
- Interaction between IC and DPV
- Staff requested GEH to address several operational performance issues
- Open items related to materials qualification and ISI

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 5 – SER Sections 5.4.7 and 5.4.8

Reactor Water Cleanup System

- Reactor water cleanup system is used for shutdown cooling (residual heat removal)
- Open Item related to thermal mixing and decay heat removal capacity pending
- RWCU designed for full reactor pressure – no intersystem LOCA concerns
- Water chemistry will be maintained according to RG 1.56 "Maintenance of Water Purity in Boiling Water Reactors" and the EPRI report series "BWR Water Chemistry Guidelines"

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 5 – SER Section 5.4.12

Reactor Coolant System High Point Vents

- RPV vent line continuously open to main steam line
- Non-condensable gas removed through off-gas system
- Additional vent opened during refueling
- RPV vent system is not part of ECCS
- No Open Items
- No COL Action Items

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 5

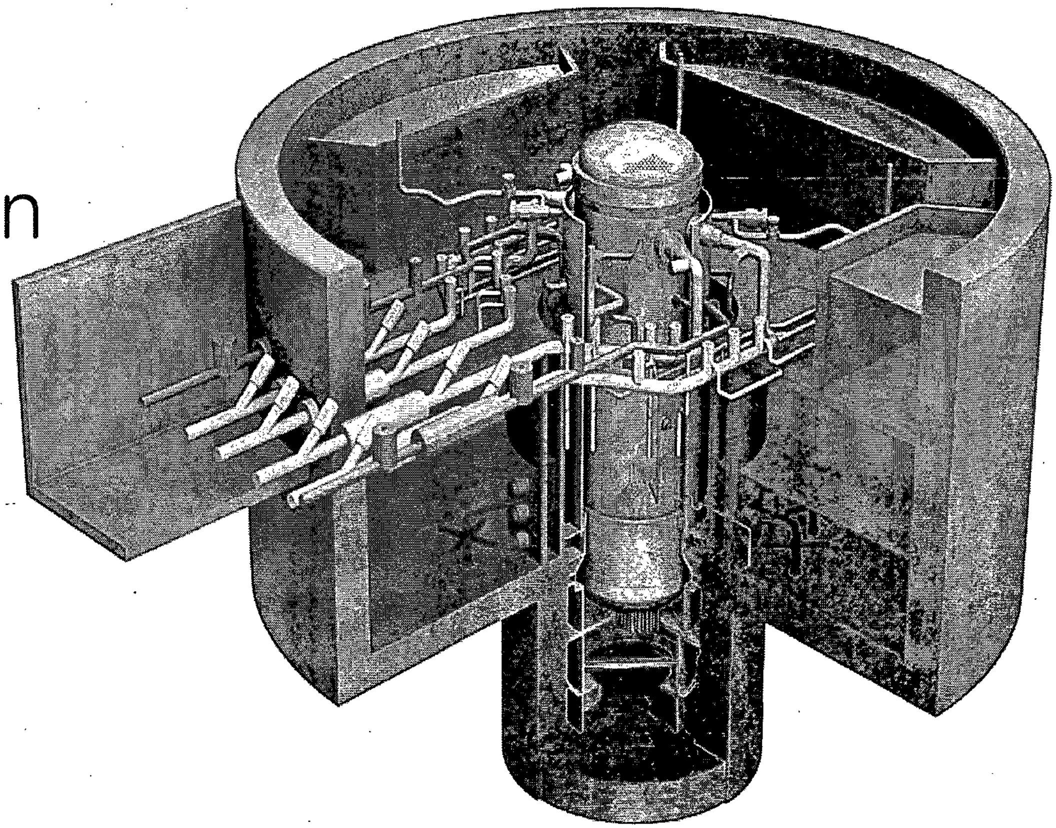
Conclusions

- Considerable progress has been made towards resolving the staffs requests for additional information (RAIs)
- Staff continues to engage GEH in discussions to resolve open items and any additional RAIs
- Although the staff believes that GEH is making progress towards an acceptable design, based on the open items that remain to be resolved the staff is unable to finalize its conclusions regarding the reactor coolant system and connected systems

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 5

Discussion/Committee Questions

ESBWR DCD Chapter 5
Reactor Coolant System and Connected Systems



Advisory Committee on Reactor Safeguards

Jerry Deaver
Joel Melito
Jeff Waal

October 25, 2007

GE Hitachi Nuclear Energy

Presentation Content

- Chapter 5 Overview
- Section Descriptions
- Summary

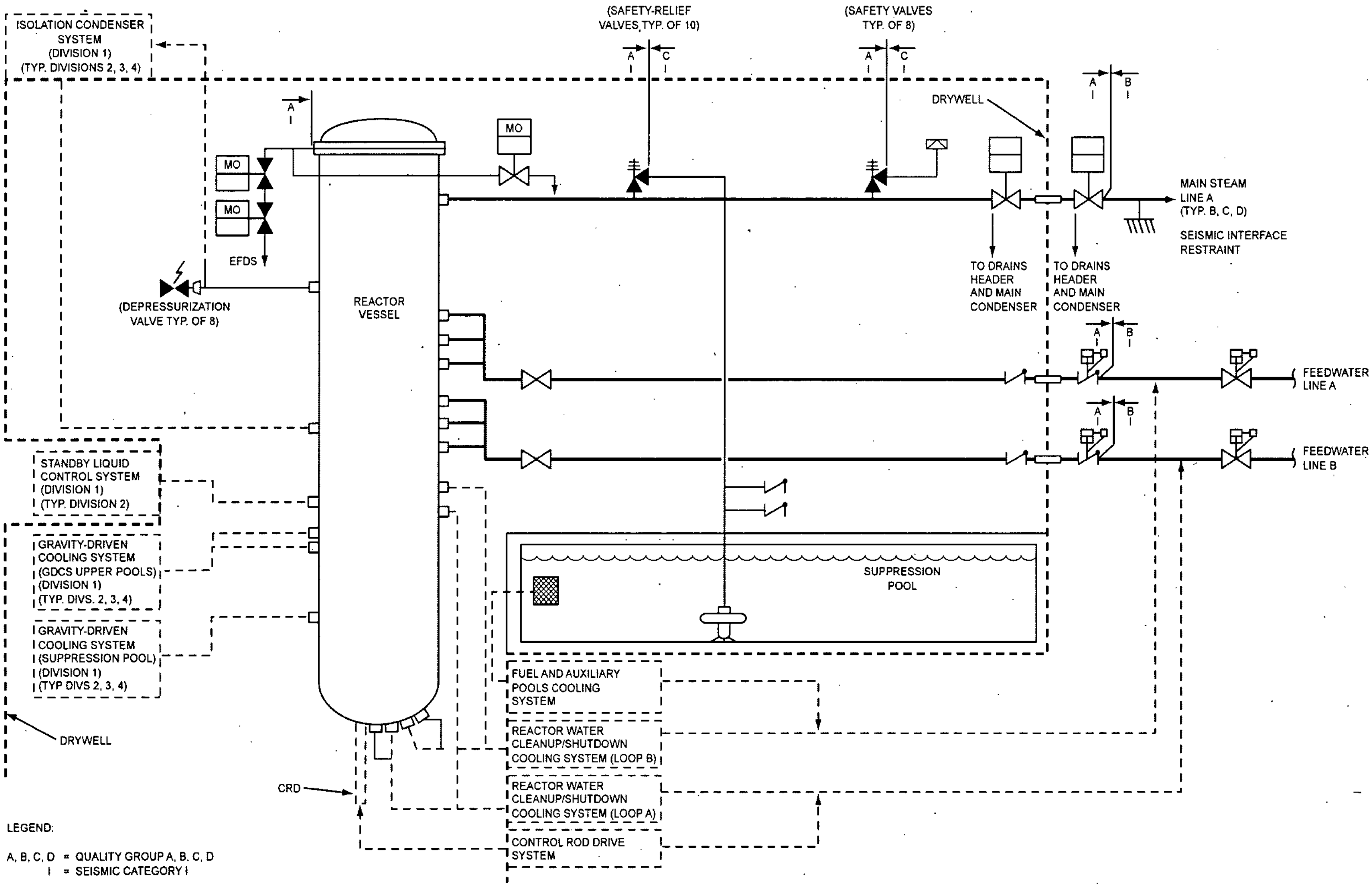
Chapter 5 Overview

- 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

Section 5.1 - Summary Description

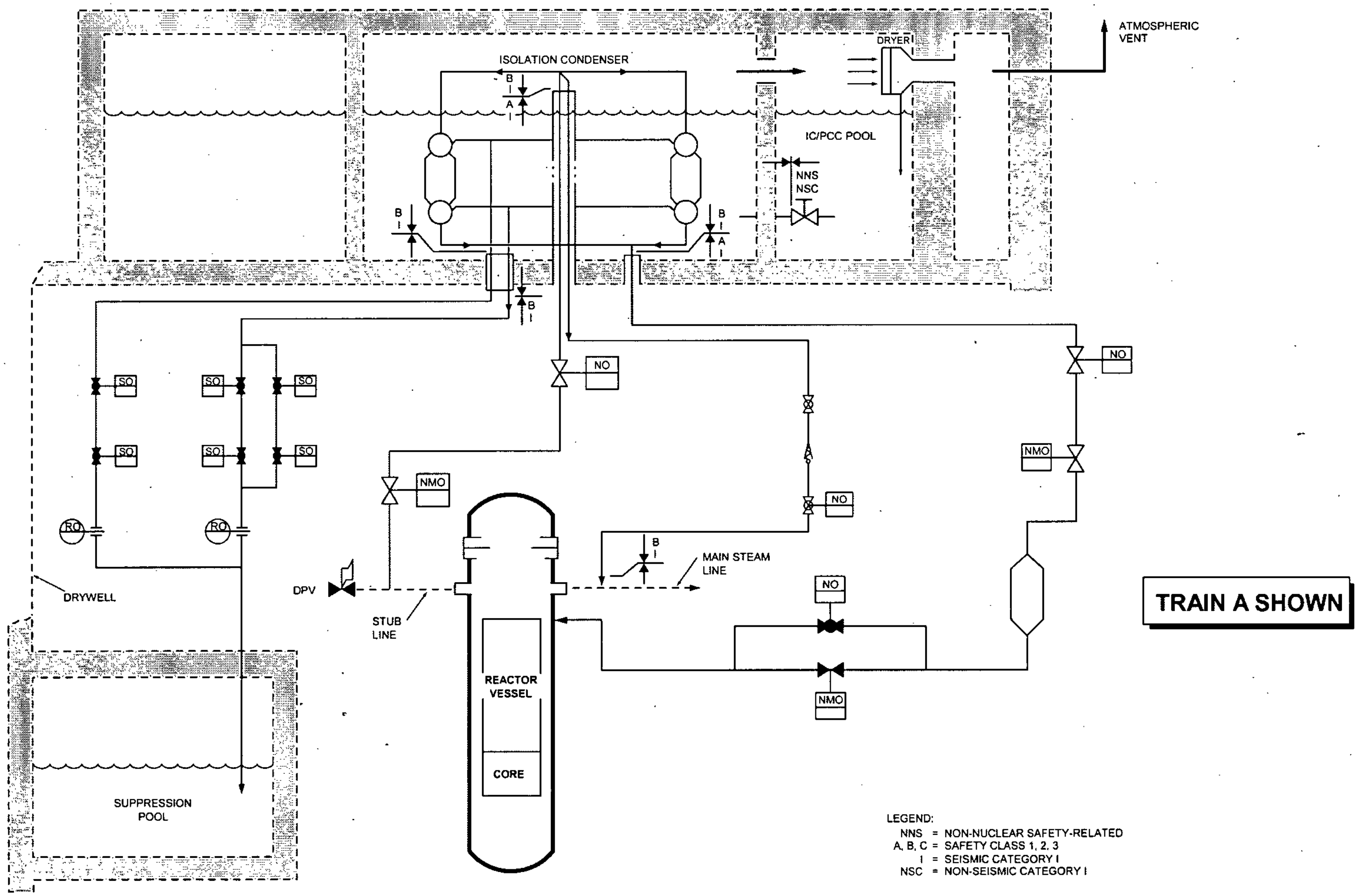
- Section 5.1 provides summary level discussion of RCS and connected systems

Section 5.1 - Summary Description



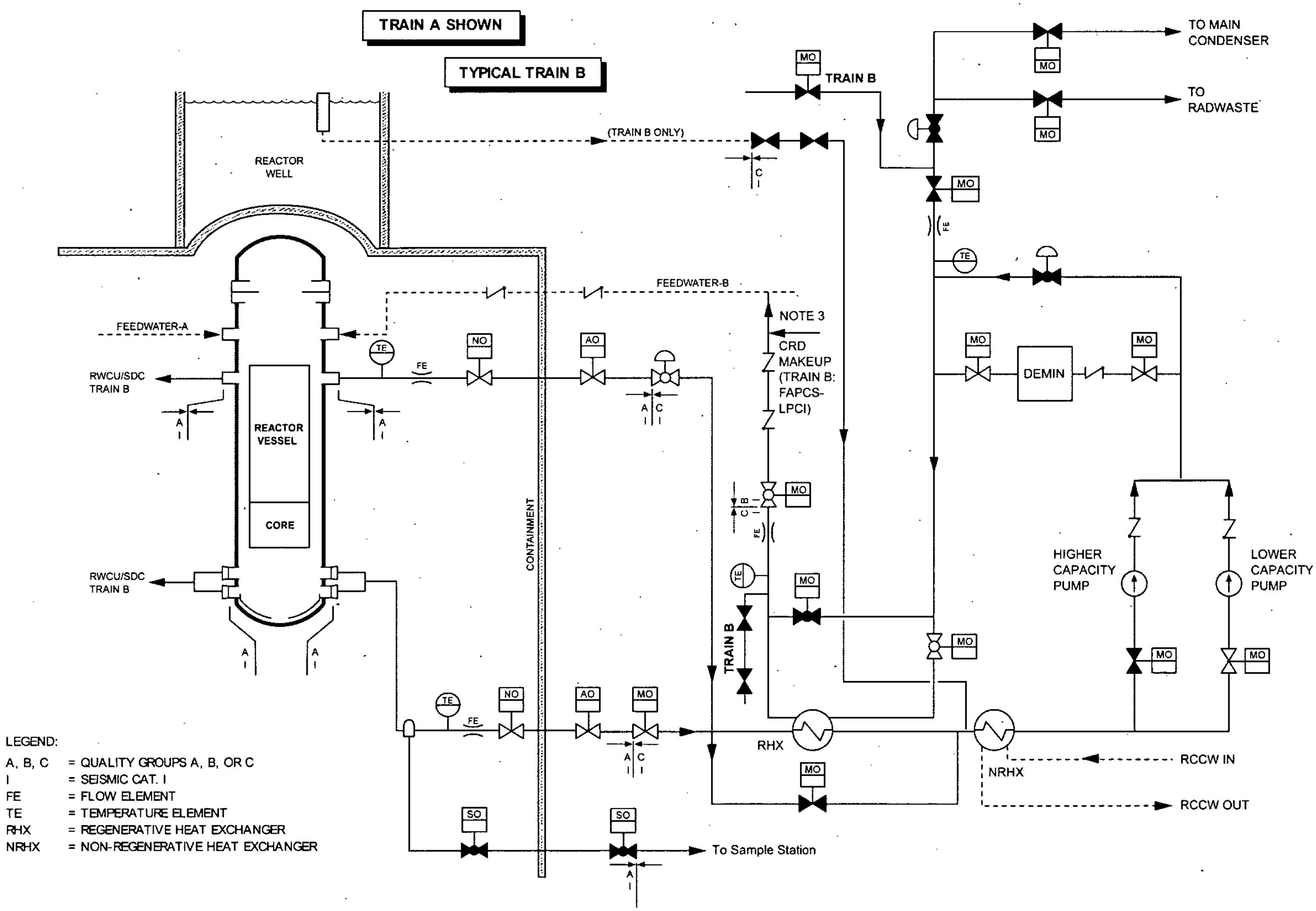
Nuclear Boiler System

Section 5.1 - Summary Description



Isolation Condenser System

Section 5.1 - Summary Description

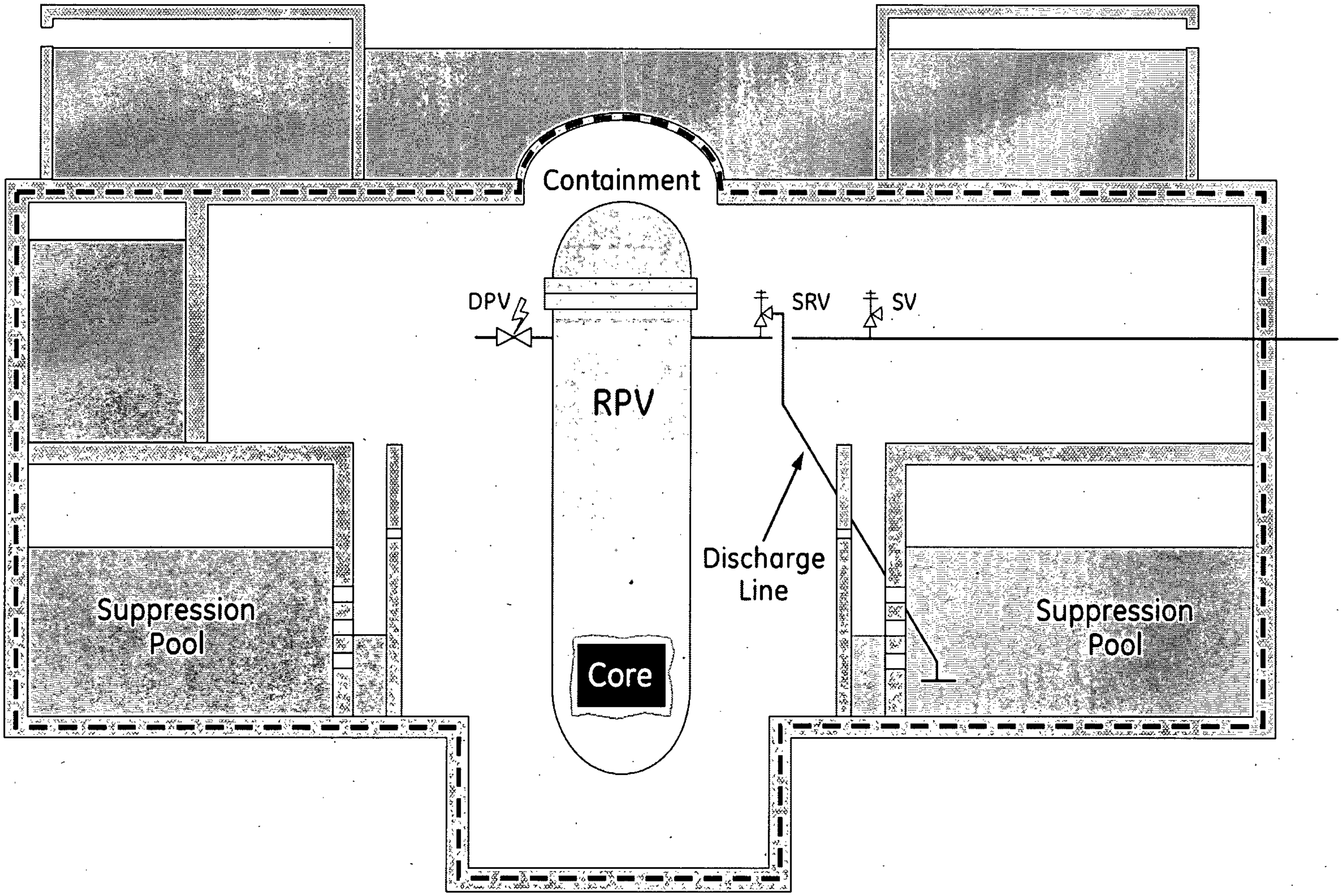


RWCU/SDC System

Section 5.2 – Integrity of RCPB

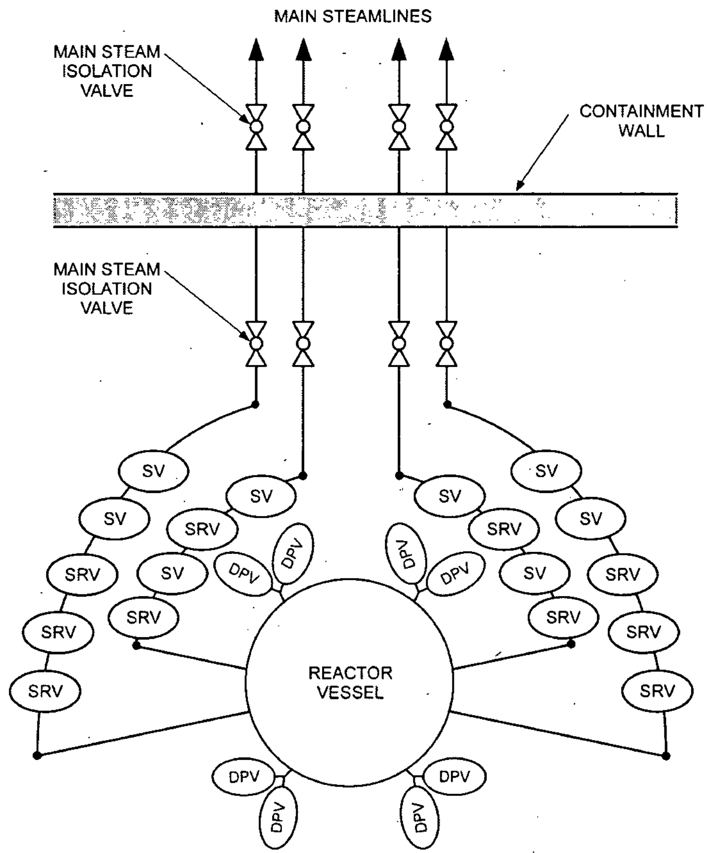
- Section 5.2 provides measures employed to provide and maintain the integrity of the RCPB that includes:
  - > Compliance with Codes and Code Cases
  - > Overpressure Protection
  - > RCPB Materials
  - > Preservice and Inservice Inspection and Testing of RCPB
  - > RCPB Leakage Detection

Section 5.2 – Integrity of RCPB



Safety Relief Valve Schematic Elevation

Section 5.2 – Integrity of RCPB



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

Section 5.2 – Integrity of RCPB

Reactor Coolant Pressure Boundary Material (typical)

- Piping, Tubular Products:
  - Steam – CS, seamless, SA 333 Gr 6
  - Feedwater – Low Alloy, seamless, SA 335 Gr P22
  - Condensate – Stn Stl, seamless, SA 213 or SA 312 Tp 316L
- Fittings, Flanges (forged):
  - Low Alloy – SA 336 Gr F22
  - Carbon Steel – SA 350 Gr LF2, SA 508 Gr 1 or SA 420 Gr WPL-6
  - Stainless Steel – SA 182 or SA 336, and SA 403 fittings, Grades 304, 304L, 316 or 316 L
- Isolation Condenser Tubing: Modified Inconel Alloy 600

Section 5.2 – Integrity of RCPB

Reactor Coolant Pressure Boundary Material (typical)

- Valves:
  - Carbon Stl – SA 216 Gr WCA & WCB, SA 352 Gr LCB, SA 105
  - Low Alloy Stl – SA 217 Gr WC9, SA 352 Gr LC1, SA 182 or SA 336 – Gr F21 or F22
  - Stainless Stl – SA 217 Gr CA15, SA 351 Gr CF8M, SA 182 Gr F6a, SA 336 Gr. F6
- Reactor Pressure Vessel
  - Shells & Heads – plate – SA 533 Gr 3 Cl 1
    forging – SA 508 Gr 3 Cl 1
  - Nozzles – forged, low alloy – SA 508 Gr 3 Cl 1
    forged, CS – SA 350 Gr LF2 or SA 508 Gr 1

Section 5.2 – Integrity of RCPB

Measures to Avoid Stress Corrosion Cracking in Stn Stl

- Avoidance of sensitization
  - Materials supplied in solution heat treated 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

Section 5.2 – Integrity of RCPB

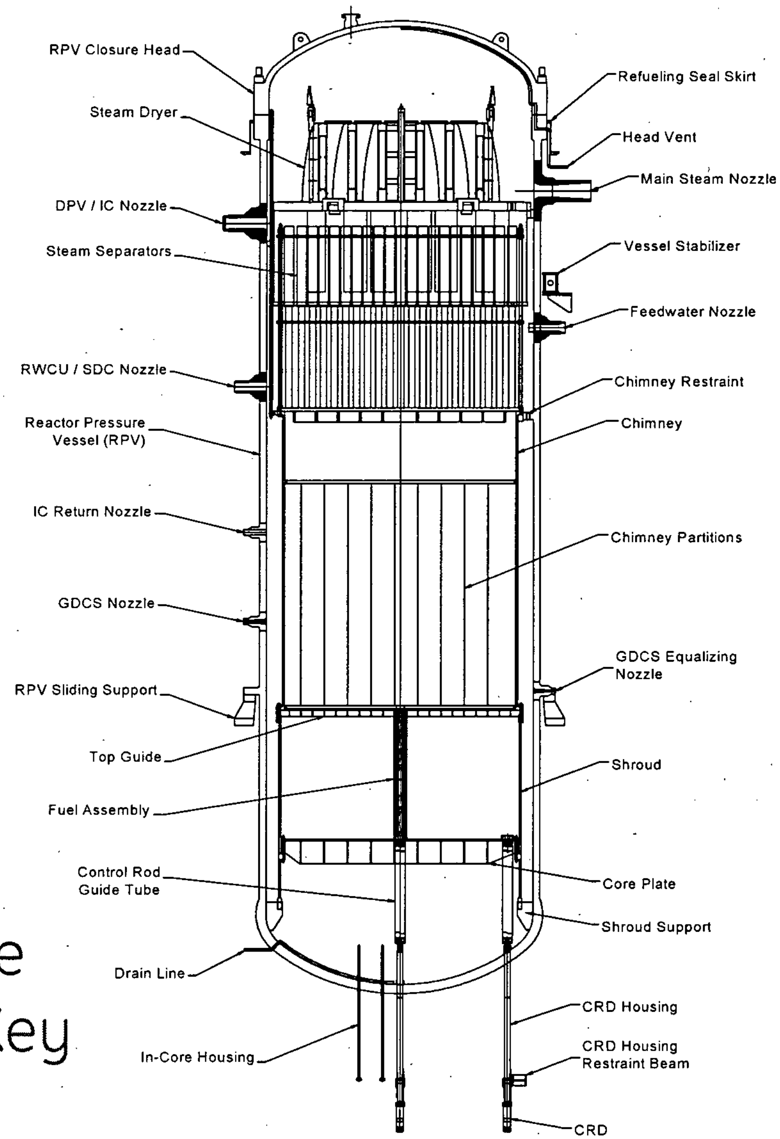
LD&IS Monitoring - Automatic Isolations

- Drywell pressure
- ICS steam/condensate flow
- Main steam line flow
- RWCU/SDC flow
- MSL tunnel temperature
- ICS radiation
- MSL pressure
- Condenser vacuum

Section 5.3 – Reactor Vessel

- Section 5.3 provides description of the reactor vessel that includes discussion of:
  - >Reactor Vessel Materials & Processes
  - >Pressure / Temperature Limits
    - \- Based on 10CFR50 App. G and RG 1.99
  - >Reactor Vessel Integrity
    - \- Proven Materials & Fabrication Methods
    - \- NDE Inspections
    - \- Inservice Surveillance Program

Section 5.3 – Reactor Vessel



Reactor Pressure Vessel System Key Features

Section 5.3 – Reactor Vessel

ESBWR RPV Changes from Prior BWRs

- RPV Supported by 8 Sliding Supports
- 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
- 6 Large Ring Forgings Are Used; Closure Flanges and 4 Lower Vessel Forgings Including Core Beltline Region
- RPV Head Vent Exits from Main RPV Body
- Main Steam Flow Restrictor Is Integral With RPV Nozzle

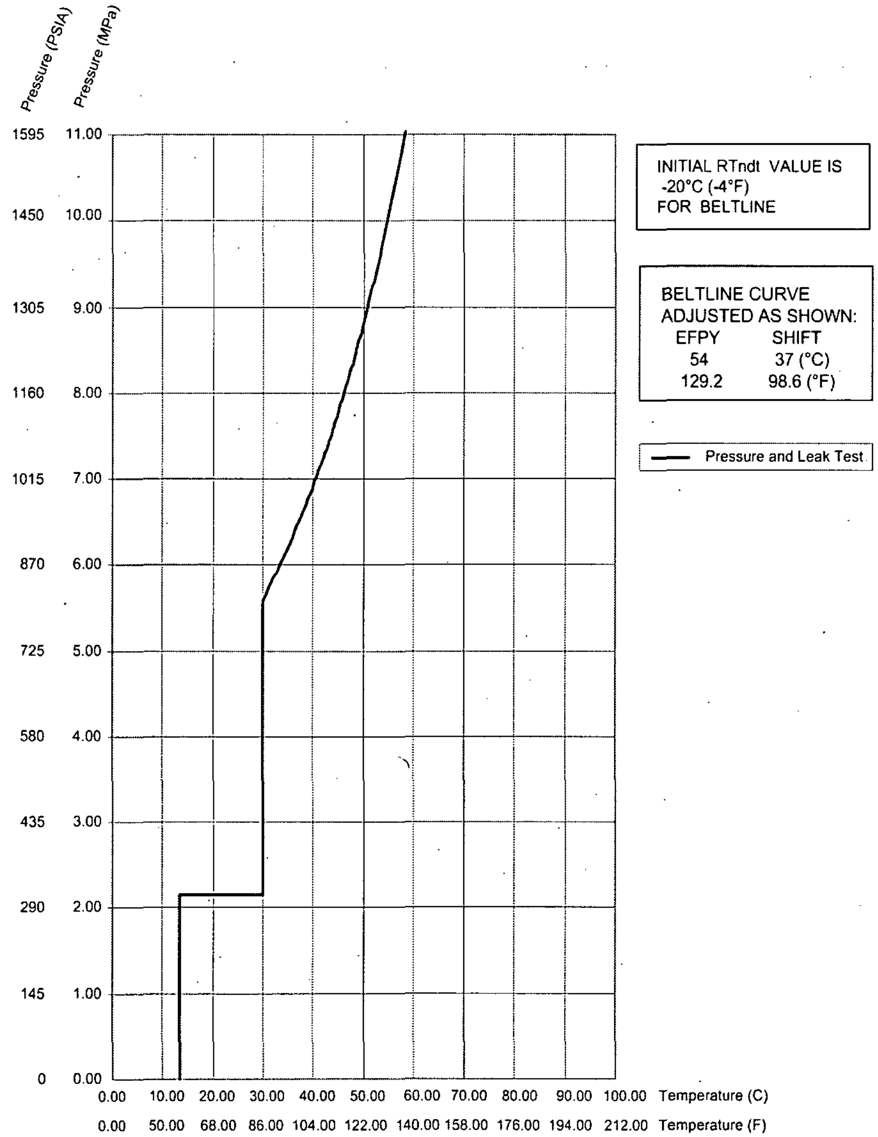
Section 5.3 – Reactor Vessel

RPV Materials & Process Controls

- Designed & Fabricated in Accordance With ASME Section III, Subsection NB, Class 1 Requirements
- Max. Limits on Copper, Phosphorus, and Nickel on RPV Materials
- Post Weld Heat Treat Temperatures 593°C Min.; 635°C Max.
- Initial Rt_{NDT} = -20°C for All Materials

Section 5.3 – Reactor Vessel

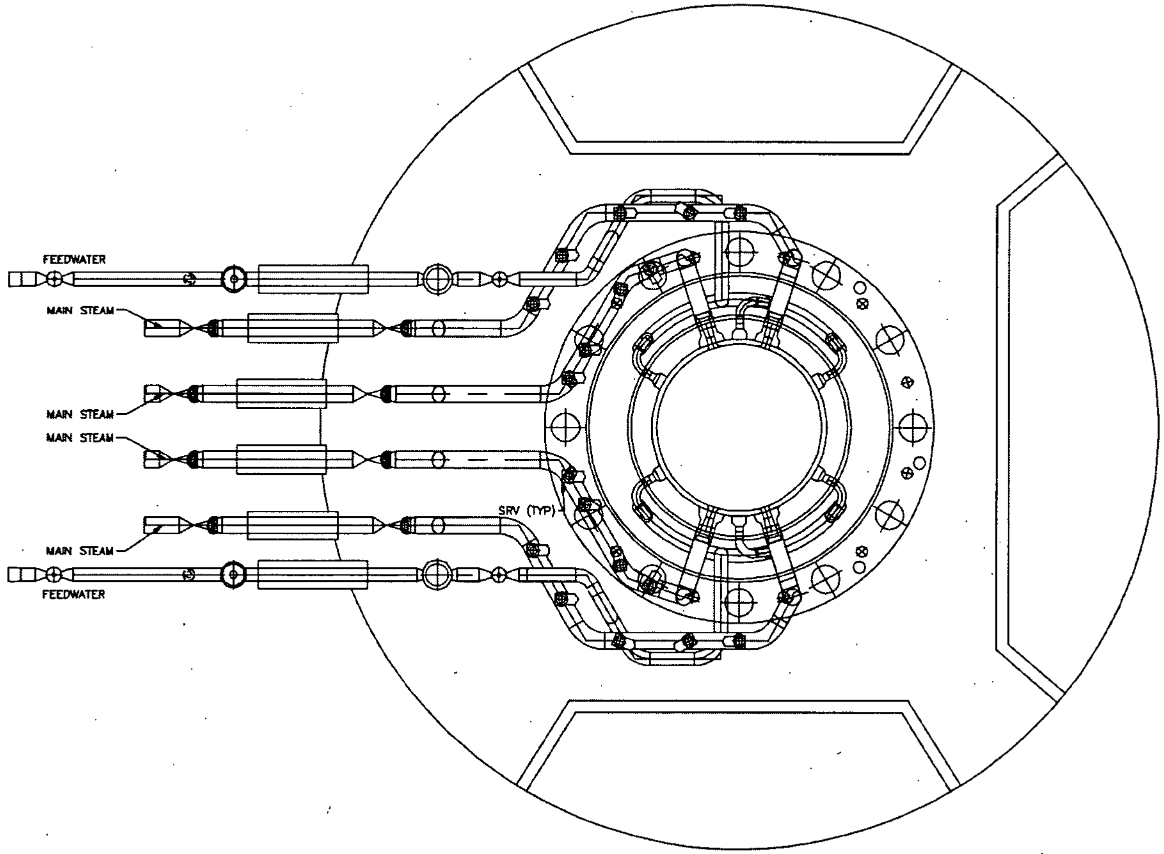
Minimum Temperature vs. Pressure Curve for Hydrotest (Representative Curve)



Section 5.4 – Component and Subsystem Design

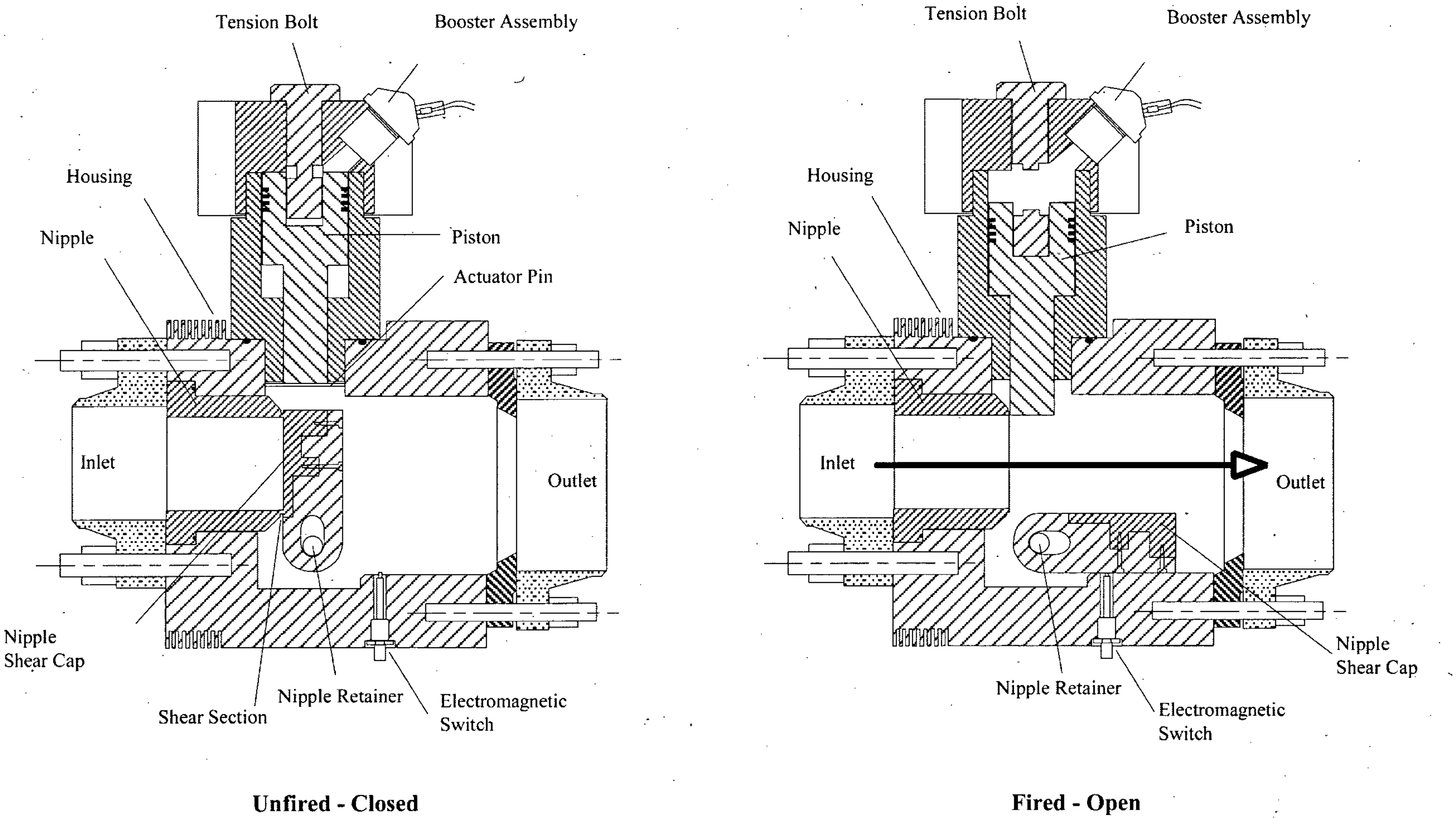
- Section 5.4 provides description of the RCS component and subsystem design that includes discussion of:
  - > **Reactor Recirculation System (Natural Circulation)**
  - > **Reactor Coolant Piping**
  - > **Main Steamline Flow Restrictors – integral with Main Steam Nozzle**
  - > **Main Steamline Isolation System - MSIVs**
  - > Isolation Condenser System
  - > Reactor Water Cleanup/Shutdown Cooling System
  - > **Main Steamlines and Feedwater Piping**
  - > Reactor Coolant System High Point Vents
  - > **Safety and Relief Valves and Depressurization Valves**
  - > Component Supports

Section 5.4 – Component and Subsystem Design



Layout of Main Steam and Feedwater Lines

Section 5.4 – Component and Subsystem Design

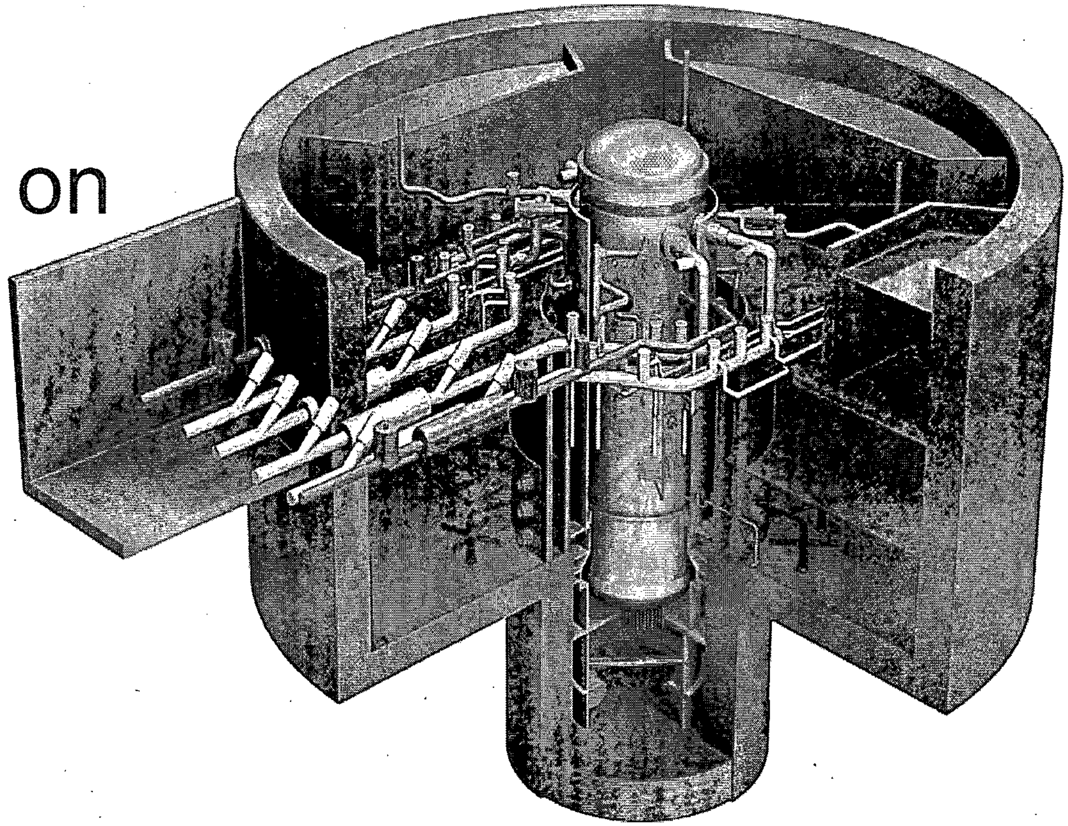


NBS Depressurization Valve

Summary

- Chapter 5 Provides Sound Description of ESBWR Reactor Coolant System and Connected Systems
- GEH Is Working With the NRC Staff to Address Remaining Open Items

ESBWR DCD Chapter 12
Radiation Protection



Advisory Committee on
Reactor Safeguards

Erik Kirstein
Frostie White
October 25, 2007

GE-Hitachi Nuclear Energy

Presentation Content

- Chapter 12 Overview
- Design Parameters
- Applicable References
- Summary

Chapter 12 Overview

- Chapter 12 Administrative programs and procedures, in conjunction with facility design, ensure that the occupational radiation exposure to personnel will be kept as low as reasonably achievable (ALARA)
- ALARA
- Radiation Sources
- Radiation Protection
- Dose Assessment
- Health Physics
- Minimization of Contamination and Waste Generation

Chapter 12 Overview (cont.)

An applicant referencing the ESBWR DCD will:

- Demonstrate compliance with Regulatory Guides 1.8, 8.8 and 8.10.
- Provide criteria and/or conditions under which various operating procedures and techniques are employed to ensure occupational radiation exposures are ALARA using the guidance of NUREG-1736.
- Ensuring offsite doses for liquid and airborne effluents and doses comply with the applicable sections of 10CFR50, Appendix I; 10CFR20, Appendix B, 10CFR Parts 20.1301 and 20.1302, respectively.

Chapter 12 Overview (cont.)

An applicant referencing the ESBWR DCD will:

- Provide procedures for operation, calibration of area radiation monitors, and placement of portable area radiation monitors.
- Provide a description of the operational radiation protection program.
- Provide a description of plant health physics equipment, instrumentation and facilities.
- Provide a description of the portable instruments that accurately measure radio-iodine concentrations in plant areas under accident conditions, and the training and procedures on use of these instruments.

Design Parameters

Requirements of design parameters for standard design are contained in:

- 10CFR Part 20
- 10CFR Part 50.34(f)(xxvii)
- 10CFR Part 50.36a
- 10CFR Part 50.68(b)(6)

Design Parameters – Ensuring Occupational Doses are ALARA – DCD Section 12.1

- General design consideration for ALARA exposures (minimize time and radiation levels).
- Equipment design consideration for ALARA exposures (accessibility, maintenance facilitation, equipment materials).
- Facility layout design considerations for maintaining exposures ALARA (equipment location, performing equipment service in lower radiation area, providing adequate space for moveable shielding during plant operation activities).

Design Parameters – Radiation Sources – DCD Section 12.2

- Reactor Vessel core sources for flux and gamma spectra.
- Equipment and system sources.
- Airborne effluent releases and doses offsite in accordance with 10CFR Part 50, Appendix I.
- Liquid effluent releases and doses offsite in accordance with 10CFR Part 50, Appendix I.
- Onsite airborne sources during normal operation and refueling.

Design Parameters – Radiation Protection Design – DCD Section 12.3

- Radiation zoning
- Radiation shielding
- Ventilation
- Area Radiation and Airborne Radioactivity Monitoring Instrumentation for normal, AOO and accident conditions.
- Post-Accident access requirements such as access/egress routes, operator actions and control points.
- Post-Accident radiation zone maps utilizing highest expected dose.

Design Parameters – Dose Assessment – DCD Section 12.4

Highest expected doses assessed for the following activities:

- Drywell dose for MSIV repair, SRV maintenance and testing, FMCRD work and maintenance and ISI.
- Reactor Building dose for Reactor Pressure Vessel access/reassembly, refueling and CRD HCU work.
- Fuel Building dose during refueling activities.
- Turbine Building dose for turbine overhaul and condensate treatment.
- Radwaste Building dose for equipment maintenance, shipment handling and radwaste processing.
- Work at power dose for HP coverage, surveillance activities and minor equipment repair.

Design Parameters – Operational Radiation Protection Program – DCD Section 12.5

- Health Physics facilities located in Service Building.
- COL applicant provides detailed description of HP equipment, instrumentation and facilities.
- COL applicant provides a description of operational radiation protection program.

Design Parameters – Minimization of Contamination – DCD Section 12.6

- Contamination minimization through design features, such as stainless steel lined equipment and sumps, skid-mounted radwaste systems, Spent Fuel Pool liner and leak detection system and no concrete block shield wall construction, to facilitate decommissioning and minimize waste generation.
- Design features that minimize generation of waste such as LWMS and SWMS process streams and waste segregation.

Applicable References

- NRC NUREG – 0800, Standard Review Plan in effect as of December 1, 2005
- Regulatory Guides – 1.69, 1.8, 1.97, 1.109,1.111, 1.113, 8.2, 8.8, and 8.10
- NUREG-1736
- NUREG-0737, Items II.F.1 and III.D.3.3.
- NUREG/CR-4013
- NUREG/CR-4653

Summary

Confirmatory/Open Items:

DCD Section 12.1 - Ensuring Occupational Dose ALARA – 1 open item – Provide justification for burnup value used in shielding analysis.

DCD Section 12.2 - Radiation Sources – 5 open items and 2 confirmatory

- Some radionuclides for airborne and liquid release rates and source terms could not be confirmed by the Staff.
- Resolve source term assumptions.
- Some liquid and airborne releases could not be confirmed by the Staff.
- Provide a cost –benefit analysis.
- Justify gas effluent source term adjustment factor.

Summary

DCD Section 12.3 - Radiation Protection – 10 open items and 4 confirmatory items

- Resolve discrepancies in zone designations.
- Add COL applicant item for operational shielding features.
- Inclined Fuel Transfer Tube shielding and maximum dose rate from spent fuel assembly.
- Reactor building HVAC filter unit figures, expected dose rates in filter and adjacent rooms, description of filter change out expected, if any, under accident conditions, including dose to perform the task, and modify post-accident radiation zones for control building emergency filter units.
- Indicate which Area Radiation Monitors are required for accidents, which RG 1.97 category and revision applies, applicable range, and for those not required for accident, demonstrate compliance to ANSI/ANS-6.8.1.
- Provide post-accident radiation zone drawings based on highest expected dose values, systems/components that contain post-accident materials outside containment, access areas required to mitigate consequences of an accident, and expected personnel actions and associated doses.

Summary

DCD Section 12.4 - Dose Assessment – 3 open items – Provide plant worker staff-hours and dose estimation consistent with RG 8.19.

DCD Section 12.5 - Operational Radiation Protection Program – 2 open items

- Provide plant layout drawings for Service Building and HP offices, control points, and access/egress routes.
- Describe sources for radioactivity analysis and instrument calibration rooms, and the shielding employed for these rooms.

DCD Section 12.6 - Minimization of Contamination – 3 open items

- Describe decommissioning facility techniques.
- Describe how wet/solid waste segregation facilitates decommissioning and how ESBWR design minimizes waste generation.
- Describe how the radioactive waste tunnel is designed to minimize and mitigate spills/leaks, features to detect leakage, whether the SFP has a double liner, and whether any contaminated piping is buried.

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Presentation to the ACRS Subcommittee

ESBWR Design Certification Review
Chapter 11
"Radioactive Waste Management"

October 25, 2007

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 11

Purpose

- Brief the Subcommittee on the staff's review of Chapter 11 of the ESBWR DCD application
- Review based on applicant DCD Rev. 3 and RAI responses received from applicant
- Answer the Committee's questions

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 11

Project and Technical Review Team

- Lead PM
  - Andrea Johnson, Project Manager
- Lead Tech. Reviewers
  - Jean-Claude Dehmel, Sr. Health Physicist
  - Jay Lee, Sr. Health Physicist
  - Chang Li, Sr. Reactor Systems Engineer
  - Hulbert Li, Electronics Engineer

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 11

Outline of Presentation

- Applicable Regulations
- RAI Status Summary
- SER Technical Topics of Interest
- Open Items
- Significant COL Action Items
- Discussion / Committee questions

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 11

Key Regulations and Review Guidance

- Part 50.34a, 50.36a, & 50.34(f)(2)
- Part 50, Appendix I Design Objectives
- Part 52.47(b)(1) and 52.80(a)
- Part 20.1301, 20.1302, 20.1406, & Appendix B to Part 20
- Part 50, Appendix A, GDC 3, 60, 61, 63, & 64
- Primary SRP Sections: 11.1, 11.2, 11.3, 11.4, & 11.5
- SRP Section Interface: 2.3, 2.4, 3.2 - 3.7, 3.8, 7.5, 9.2, 9.3, 9.4, 9.5, 10.4, 12.2.2, 13, 14, 16, & 17
- Regulatory Guides 1.11, 1.21, 1.33, 1.52, 1.97, 1.109, 1.110, 1.111, 1.112, 1.113, 1.140, 1.143, 4.15, 8.8, & 8.10
- Industry Standards: AISI, ANS, ANSI, API, ASME, ASTM, NFPA, & TEMA

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RAI Status Summary

- Original number of RAIs = 88
- Number of RAIs resolved = 85
- Number of Open Items = 3

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- Primary SER Sections
  - ➢ SER 11.1 - Source Terms (Radiological)
  - ➢ SER 11.2 - Liquid Waste Management System
  - ➢ SER 11.3 - Gaseous Waste Management System
  - ➢ SER 11.4 - Solid Waste Management System
  - ➢ SER 11.5 - Process Radiation Monitoring System

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SER 11.1 Technical Topics of Interest

- Source Terms
  - ➢ Normal operation - gaseous and liquid effluent releases in unrestricted areas

    Based on ANSI/ANS 18.1-1999, and RG 1.112, Revision 1 (March 2007)
  - ➢ Design Basis - plant equipment design and radiation shielding

    Based on 100,000 µCi/sec noble gas release at 30 minute decay (historical GE value with 1% fuel defect)

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SER 11.1 Technical Topics Reviewed

- Source Terms
  - ➢ Proper application of ANS/ANSI 18.1-1999 and RG 1.112
  - ➢ Staff RAIs focused on:
    - descriptions and inclusions of parameters used in deriving nuclide concentrations in primary coolant and steam
    - identification of normal and potential sources of effluents
    - clarification on source terms for fission, activation, and corrosion products, including noble gases
  - ➢ Staff confirmed source terms and found source terms acceptable
  - ➢ All RAIs were satisfactorily resolved
  - ➢ All RAIs are closed, no COL action items

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SER 11.2 Technical Topics of Interest

- Liquid Waste Management System
  - ➢ Equipment design for normal operations and anticipated operational occurrences
  - ➢ Features to process, collect, and treat liquid process streams, and control effluent releases
  - ➢ Design relies on mobile radwaste subsystems connected to permanently installed equipment
  - ➢ Key SRP interface: 9.3, 11.3, 11.4, 11.5, & 12.2

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SER 11.2 Technical Topics Reviewed

- Liquid Waste Management System
  - ➢ Staff RAIs focused on:
    - consistency of tanks design basis against RG 1.143
    - system flow paths, process streams, effluent discharge
    - basis for system performance (DF) in treating liquid wastes
    - scope of COL action items for mobile waste processing systems
    - ITAAC on mobile systems configuration, plant system interfaces, and operation
  - ➢ Effluent monitoring tied to COL action items in DCD Section 11.5
  - ➢ One RAI remains open mobile waste processing systems
  - ➢ DCD Chapter 11.2 identifies two COL action items

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SER 11.3 Technical Topics of Interest

- Gaseous Waste Management System
  - ➢ Equipment design for normal operations and anticipated operational occurrences
  - ➢ Features to process, collect, and treat gaseous process streams, and control effluent releases
  - ➢ Design relies on permanently installed equipment
  - ➢ Key SRP interface: 9.4, 11.2, 11.4, 11.5, and 12.2

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SER 11.3 Technical Topics Reviewed

- Gaseous Waste Management System
  - Staff RAIs focused on:
    - qualification of OGS to withstand internal explosions
    - system design features and specifications
    - basis for system performance (holding time) in treating gaseous wastes
    - scope of COL action items in defining system performance and effluent monitoring
  - One confirmatory item remains open on COL holder's QA program
  - Effluent monitoring tied to COL action items in DCD Section 11.5

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SER 11.4 Technical Topics of Interest

- Solid Waste Management System
  - Equipment design for normal operations and anticipated operational occurrences
  - Features to process, collect, and treat solid and wet wastes, and control effluent releases
  - Design relies on mobile radwaste subsystems connected to permanently installed equipment
  - Operational Program – Process Control Program
  - Key SRP interface: 9.3, 9.4, 11.2, 11.3, & 11.5

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SER 11.4 Technical Topics Reviewed

- Solid Waste Management System
  - Staff RAIs focused on:
    - consistency of design basis against RG 1.143
    - system flow paths, process streams, effluent discharge
    - methods for processing large components and spent charcoals
    - scope of COL action items for mobile waste processing systems
    - ITAAC on mobile systems configuration, plant system interfaces, and operation
  - Effluent monitoring tied to COL action items in DCD Section 11.5
  - Two RAIs remain open, ITAAC and DCD scope mobile systems
  - DCD Chapter 11.4 identifies 12 COL action items

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SER 11.5 Technical Topics of Interest

- Process Radiation Monitoring System
  - Equipment design for normal operations and anticipated operational occurrences
  - Features to characterize types and amounts of radioactivity in process streams and effluents, and control effluent releases
  - Design relies on combination of skid mounted-subsystems and permanently installed equipment
  - Operational Programs – ODCM, SREC, REMP
  - Key SRP interface: 7.5, 11.2, 11.3, & 11.4

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SER 11.5 Technical Topics Reviewed

- Process Radiation Monitoring System
  - Staff RAIs focused on:
    - design basis against SRP 7.5 and 11.5 and RG 1.21 and 4.15
    - instrumentation systems, sampled streams, effluent discharge points
    - automatic safety functions, isolation and termination of releases
    - scope of COL action items for instrumentation systems
    - operational programs, ODCM, SREC, and REMP
  - 18 confirmatory items remain open
  - DCD Chapter 11.5 identifies five COL action items

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SER Chapter 11 Open RAI Items

- RAIs are associated with status of mobile systems, DCD "conceptual" design and linked with COL action items on plant interfaces
- Resolution of open RAIs expected in the context of DCD Rev. 4 and 5 updates

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Significant SER Chapter 11 COL Action Items

- A number of COL action items are expected to change
  - COL action items related to plant and site specific features, defined at the COL stage
  - COL action items for mobile processing equipment and plant interfaces
  - COL action items for operational programs, ODCM, SREC, REMP, and PCP
- Resolution of COL action items expected in the context of DCD Rev. 4 and 5 updates

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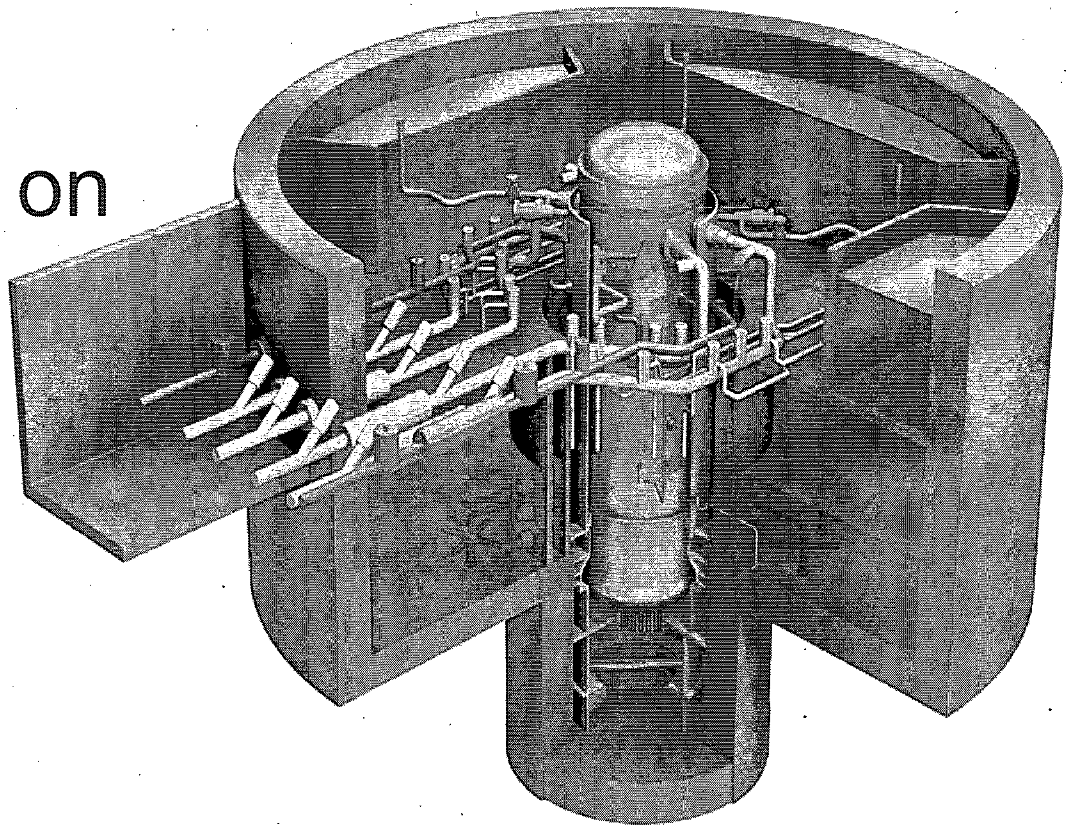
Discussion/Committee Questions

ESBWR DCD Chapter 11
Radioactive Waste Management



Advisory Committee on Reactor Safeguards

Erik Kirstein
Dale McCullough
Frostie White

October 25, 2007

GE-Hitachi Nuclear Energy

Presentation Content

- Chapter 11 Overview
- Design Parameters
- Applicable References
- Summary

Chapter 11 Overview

- 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
- Solid waste management
- Process and effluent monitoring and sampling

Chapter 11 Overview (cont.)

An applicant referencing the ESBWR DCD will:

- Ensure that LWMS mobile/portable equipment operation and testing comply with RG 1.143.
- Identify mobile/portable LWMS and SWMS non-radioactive connections that may become radioactive due to leakage, valve errors or operating conditions in accordance with the guidance in IE Bulletin 80-10.
- Describe the implementation of operating procedures and design features for installation and operation of LWMS and SWMS mobile/portable equipment to minimize facility and environment contamination, minimize waste generation, and facilitate decommissioning.
- Develop a Process Control Program in accordance with 10CFR20, Appendix G, and classifies waste generated in accordance with 10CFR61.55 and 10CFR61.56.

Chapter 11 Overview (cont.)

An applicant referencing the ESBWR DCD will:

- Provide an overall site management plan for a temporary storage facility in accordance with SRP 11.4 if one is established.
- Establish the lower limit of detection for the specific effluent radiation monitoring subsystem based on site-specific conditions and operating characteristics.
- Develop an ODCM for calculation of gaseous and liquid effluents, planned discharge flow rates, operational and trip setpoints, and address site-specific conditions in assessing radiation exposure, including N-16 and skyshine dose in accordance with 10CFR20.1301(e) and 10CFR20.1302.
- Develop site-specific Offsite Dose Calculation for the maximally exposed offsite individual and population doses for gaseous and liquid effluents in accordance with 10CFR50, Appendix I.
- Develop instrument sensitivities for gaseous and liquid samples.

Chapter 11 Overview (cont.)

- Systems have backup capability to permit continued processing while performing maintenance.
- Radwaste systems are designed in accordance with Regulatory Guide 1.143.
- Used experience of operating BWRs in design and operational considerations to minimize contamination, minimize effluent, and reduce waste generation.
- Considered ALARA and dose minimization special design and operation features.

Design Parameters

Requirements of the design parameters for standard design are contained in:

- 10CFR Parts 20.1301(e), 20.1302, 20.1406, 20.2007, Part 20 Appendices B and G
- 10CFR Parts 50.34a, 50.34(f)(vii), 50.34(f)(xvii), 50.34(f)(xxvii), 50.34(f)(xxviii), 50.36a, Part 50 Appendix A – GDCs 3, 19, 60, 61, 63, and 64, and Part 50 Appendix I
- 10CFR Part 61, 10CFR Parts 61.55 and 61.56
- 10CFR Part 71
- 40CFR Part 190
- 49CFR Parts171 through 180

Design Parameters- Source Term – DCD Section 11.1

- Reactor coolant source term calculated using ANSI/ANS – 18.1-1999.
- Design basis noble gas release rate of 100K μCi/sec at 30 minutes decay.
- Design basis iodine source term based on I-131 leak rate of 700 μCi/sec from fuel.
- Coolant source terms used in supporting analyses for DCD Chapter 12 (equipment activity inventories) and Chapter 15 (DBA dose evaluations).

Design Parameters – Liquid Radwaste – DCD Section 11.2

- Similar to operating BWRs with collection tanks for incoming liquid waste streams (Low Conductivity, High Conductivity, Chemical Waste, and Detergent Waste).
- Processing equipment consisting of filters to remove insoluble contaminants and demineralizers or reverse osmosis to remove soluble contaminants.
- Sample tanks to collect batches of processed liquid which can be either pumped to condensate storage, reprocessed or released to the environment, depending on sample analysis and allowable release limits.
- Like operating BWRs, incorporates substantial skid-mounted mobile process systems, using best available processing technology. Reduces solid radwaste generation, maintenance and dose compared to existing designs.

Design Parameters – Liquid Radwaste contd.

- Current BWR radwaste systems typically have equipment that has been abandoned because of change in process philosophy or because the existing equipment was high maintenance/high dose. The use of mobile processing avoids this practice.
- Designed for total recycle of liquid radwaste.
- Designed for ALARA, minimizing the spread of contamination, and facilitating decommissioning.
- Operating experience utilized in the selection of processing equipment. Avoided the use of high maintenance equipment, such as evaporators.

Design Parameters – Gaseous Radwaste – DCD Section 11.3

- Gaseous radwaste is typical of existing BWR offgas systems: H_2/O_2 recombination, moisture removal, holdup and decay in charcoal beds.
- The OGS is capable of processing three times the source term without affecting delay time of the noble gases.
- Based upon a conservative analysis, the leak rates and doses are expected to be less than one-fifth of design basis.

Design Parameters – Solid Radwaste – DCD Section 11.4

- Same as existing BWRs using a process control program; wet solid waste from plant filters and demineralizers is collected in spent resin and phase separator tanks, and pumped to waste containers where it is dried to burial criteria or the waste acceptance criteria.
- Waste streams are segregated to reduce the amount of B class waste and to enable placement of B waste into containers that are loaded in shipping casks which can accommodate higher dose waste.
- SWMS is designed to meet the guidance of RG 1.143.
- The SWMS design keeps plant personnel exposure ALARA (RG 8.8) during normal operation and plant maintenance by incorporating the use of special features such as cameras.

Design Parameters – Process and Effluent Monitoring and Sampling – DCD Section 11.5

- Similar to existing BWRs, radiation monitors initiate safety-related functions, allow diagnosing plant gas and liquid waste process streams, and monitoring gas and liquid effluent streams.
- Safety-related monitors initiate closure of HVAC dampers associated with Reactor Building, Refuel Handling Area, Control Building, Fuel Building General Fuel Pool Areas upon high radiation.
- Safety-related monitors also initiate closure of drywell sumps discharge lines, isolation condenser containment isolation valve and containment purge exhaust.
- Radiation monitors and sample points provided to monitor gaseous and liquid effluent and process streams.
- Instrumentation compatible for normal, AOO and accident conditions.

Applicable References

- NUREG – 0800, Standard Review Plan as of February 2005
- NUREG – 0737, Item II.F.I
- NUREG – 0016
- Regulatory Guides – 1.21, 1.45, 1.97, 1.143, 4.15, 8.8, 8.10
- IE Bulletin – 80-10
- Generic Letter 89-01
- ANSI/ANS-55.4
- ANSI/ASME B31.3
- ANSI/ANS-40.37
- ANSI/HPS N13.1-1999
- ANSI/IEEE N42.18-1980
- ANSI/ANS-18.1-1999
- ASME Boiler and Pressure Vessel Code, Sections II, VIII - Division 1 or 2, and IX.

Summary

DCD Section 11.1 - Source Terms – no open or confirmatory items.

DCD Section 11.2 - Liquid Radwaste – five confirmatory items and one open item

DCD Section 11.3 - Gaseous Radwaste – one confirmatory item

DCD Section 11.4 - Solid Radwaste – four confirmatory items and two open items

DCD Section 11.5 - Process and Effluent Monitoring and Sampling – 18 confirmatory items

Clarifications on Chapter 11.1 Source Term Presentation
ACRS ESBWR Presentation, Oct. 25, 2007

The "1% fuel defect" means 1% of fuel rods in the reactor core experience some degree of fuel cladding defects. It is recognized that 1% fuel rod defect does not represent 1% of the core or gas gap inventory in radioactivity levels. In the context of the presentation, the slide on Chapter 11.1 should not have included this value.

In NUREG-0016, the default noble gas release rate is assumed to be 55,000 μCi/sec at 30 minute decay and normalized to 3400 MWt, based on actual measurements at operating BWRs (see NUREG-0016, page 2-11 and Table 2-6, page 2-12).

This value scales up to 73,000 μCi/sec at 30 minute decay for the ESBWR at 4500 MWt power level, which is bounded by the 100,000 μCi/sec at 30 minute decay stated in DCD Tier 2, Rev. 3, Section 11.1.

The DCD, Section 11.1, refers to the GE NEDO-10871 (March 1973) as the basis of the noble gas source term of 100,000 uCi/sec at 30 minute decay.

It is recognized that since the NEDO report was issued, fuel performance has improved and consequently, the noble gas source term has decreased based on data reported in gaseous effluent release reports. Given that the ESBWR has a larger number of fuel rods, the noble gas source term is mitigated by improved fuel performance, thereby, resulting in a lower noble gas source term even though the ESBWR has a larger number of fuel rods when compared to the plants on which the noble gas source term is based.

Clarification on Chapter 11.2 & 11.3 Radioactive Waste Management Systems
ACRS ESBWR Presentation, Oct. 25, 2007

Regarding the definition of AOO (anticipated operational occurrences), NUREG-0016 defines the term as "unplanned releases of radioactive materials from miscellaneous actions, such as equipment failures, operator errors, that are not of consequence to be considered an accident." The definition does not present a list or catalog of events that would be considered AOOs.