Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title: Advisory Committee on Reactor Safeguards BWRX-300 Subcommittee: Open Session

Docket Number: (n/a)

Location: teleconference

Date: Friday, March 18, 2022

Work Order No.: NRC-1888 Pages 1-80

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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BWRX-300 GE-HITACHI TOPICAL REPORT, "BWRX-300 CONTAINMENT EVALUATION METHOD" SUBCOMMITTEE

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OPEN SESSION

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FRIDAY

MARCH 18, 2022

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The Subcommittee met via Video Teleconference, at 1:00 p.m. EDT, Jose March-Leuba, Chairman, presiding.

COMMITTEE MEMBERS:

JOSE MARCH-LEUBA, Chair

RONALD G. BALLINGER, Member

CHARLES H. BROWN, JR. Member

VESNA DIMITRIJEVIC, Member

GREG HALNON, Member

DAVID PETTI, Member

JOY L. REMPE, Member

ACRS CONSULTANT:

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

MICHAEL SNODDERLY

ALSO PRESENT:

SHAWN CAMPBELL, RES

JESUS DIAZ-QUIROZ, GEH

MICHAEL DUDEK, NRR

THOMAS GEORGE, Zachary Engineering

BERNARD GILLIGAN, GEH

SYED HAIDER, NRR

ROSEANNE HARRINGTON, GEH

DAVID HINDS, GEH

CHARLES HECK, GEH

ANDREW IRELAND, NRR

SCOTT KREPEL, NRR

NECDET KURUL, GEH

PETER LIEN, NRR

SHANLAI LU, NRR

JUN MATSUMOTO, GEH

RYAN NOLAN, NRR

DAN PAPPONE, GEH

LISA SCHICHLEIN, GEH

JAMES SHEA, NRR

JOE STAUDENMEIER, RES

CARL THURSTON, NRR

GEORGE WADKINS, GEH

FROSTIE WHITE, GEH

CONTENT

P R O C E E D I N G S

(1:00 p.m.)

CHAIR MARCH-LEUBA: The meeting will now come to order.

This is a meeting on the BWRX-300 ACRS Subcommittee. I am Jose March-Leuba, the SC chairman.

Because of COVID-19 concerns, this meeting is being conducted in a highly minded manner. In addition to in-person attendance at NRC headquarters, the meeting is broadcast via MS Meetings. Members in attendance are Ron Ballinger, Charles Brown, David Petti. And I haven't checked on the -- Vesna, are you, are you in?

MEMBER DIMITRIJEVIC: Yes, I am.

CHAIR MARCH-LEUBA: Vesna Dimitrijevic is present, too.

And we have Steve Schultz, our consultant.

Anybody I have missed?

MR. BROWN: Is Greg going to be in the meeting?

CHAIR MARCH-LEUBA: When Greg shows up I will put it on the record.

MR. BROWN: And Joy?

CHAIR MARCH-LEUBA: Today's topic is Licensing Topic Report NEDC-33922P by General

Electric-Hitachi, BWRX-300 Containment Evaluation Method. Portions of our meeting will be closed to the public to protect GEH proprietary information.

We shall now receive the questions and provide comments plus an opportunity for spur-of-the-moment public comments before the beginning of the closed section of the meeting.

The ACRS was established by statute and is governed by the Federal Advisory Committee Act, FACA. As such, the committee can only speak to this published letter report.

The rules for participation in all ACRS meetings were announced in the Federal Register on June 13th, 2019. The ACRS section of the U.S. NRC public website provides our charter, bylaws, agendas, letter reports, and full transcripts for the open portions and all full and subcommittee meetings, including the slides presented there.

The representative federal official today is Mike Snodderly.

A transcript of the meeting is being kept. So, speak into the microphones clearly, and state your name for the benefit of the court reporter, especially during the meeting using the bridge line. Please leave the microphone on mute when not being used and when you are speaking you have 15 minutes.

I note that members Greg Halnon and Joy Rempe have joined us.

We are expecting to analyze this topic during the full committee meeting on April 6th, and possibly to write a letter.

At this point let's request Mr. Mike Dudek from NRC to present his opening remarks.

Mike, are you ready?

MR. DUDEK: Yes, sir.

So, thank you, Subcommittee Chairman March-Leuba. And thank you to the rest of the ACRS subcommittee for allowing us to present this second important topical report for GEH's BWRX-300 technology today. Specifically, this one that we're discussing this afternoon, as Subcommittee Chairman March-Leuba said, is the BWRX-300 containment evaluation method topical reports.

It was first submitted to the NRC in the fall of 2020, and subsequently had three different revisions submitted, up to and including January 7th, 2022. And the purpose of the topical report is to really present the acceptable analysis methods for the BWRX-300 containment and thermal hydraulic performance, and demonstrate that the containment designs satisfy the acceptance criteria outlined in the topical report.

And more specifically, you know, associated with that containment performance analysis, it really outlines subject topic areas such as AOOs, station blackouts, ATWS, Large and Small Break LOCAs, and things of that nature.

The staff conducted a very in-depth review on this topic report over a 2-year period, and developed a very inclusive safety evaluation reports in accordance with several rounds of RAIs and a very in-depth and topic-specific audit that was discussed.

So, without any further ado, I will turn it over back to you, Subcommittee Chairman March-Leuba, and I'll look forward to the discussions and any questions that you have today.

CHAIR MARCH-LEUBA: Thank you.

We will, without any wait, we'll go to the GE. I understand management wants to make, George Wadkins will make some introductory remarks and you will present your team.

I remind everybody we are in the open portion of the meeting. And this topical report contains proprietary information. So, reserve your questions for an hour from now when we move to the closed session.

George, go ahead.

MR. WADKINS: Thank you. My name is George Wadkins. I am the Vice President of New Power Plants and Products Licensing for GE-Hitachi.

I wish to thank the ACRS subcommittee for allowing us to present the overview of the content for this licensing topical report. Today we will be describing the use of GOTHIC thermal hydraulics code as the methodology to be used to verify that design requirements for the BWRX-300 dry containment will be met in the preliminary and final design of the BWRX-300.

This will include, for example, maximum allowed containment, post-accident pressure and temperature response.

As noted in our previous discussions with the ACRS, the BWRX-300 builds upon our most recent experiences in development and certification of the Economic Simplified Boiling Water Reactor, or ESBWR.

One major difference between the ESBWR and the BWRX-300 is in the functional design of the containment. The ESBWR utilized a compartmentalized containment with a dry well surrounding the reactor and connected systems, including piping and vales, and a wet well containing a suppression pool. The suppression pool functions to condense steam from the reactor, depressurization, and to allow passive injection of water to maintain water inventory in cooling of the reactor core, while maintaining acceptably low containment pressures and temperatures.

The BWRX-300 has a relatively large, relatively large dry containment that does not require a wet well as a containment pressure and temperature suppression feature, but it does include a passive containment cooling system for heat removal from containment.

Reactor depressurization is achieved through use of the isolation condenser system instead of through an automatic depressurization system that releases steam to the suppression pool.

And the BWRX-300 does not require the more complicated passive injection systems used in the ESBWR design.

Because of the simplified design of the containment, the analytical methods requirements are also simplified, with the application of a different methodology using GOTHIC instead of the traditional TRACG code used for containment pressure and temperature response for the ESBWR and other BWRs.

This licensing topical report seeks approval for use of GOTHIC as an acceptable code methodology for the BWRX-300.

I first want to thank the NRC staff for their in-depth review of this licensing topical report. The interactions with the NRC's technical staff, including the conduct of the audit, really helped to ensure that the content of the licensing topical report was complete, understandable, accurate, and met the applicable regulatory requirements and guidance.

I also wish to thank staff of the Canadian Nuclear Safety Commission for their participation in the joint review of this licensing topical report with the NRC staff.

And I look forward to continuing this and future interactions with the NRC staff, Canadian Nuclear Safety Commission staff, and the ACRS.

Next slide, please.

So, for this meeting we are going to provide an extensive open session discussion of the content of this licensing topical report, as shown in this agenda. This report also has a proprietary version which will be discussed in the closed session later, following this open session meeting.

During our presentation we will pause at the end of each slide to allow for questions from the ACRS members. But please feel free to raise questions at any time. If the discussions involve proprietary information, then we will request tabling the questions until the latter closed session.

Next slide, please.

I will now turn over the presentation to Lisa Schichlein, U.S. Licensing Manager.

Thank you.

MS. SCHICHLEIN: Good afternoon. My name is Lisa Schichlein, and I am the U.S. Licensing Manager for New Power Plants and Products at GE-Hitachi Nuclear Energy Americas. I would like to thank the ACRS subcommittee for the opportunity to present the BWRX-300 containment evaluation method licensing topical report.

With me on the call today are -- is the licensing engineer Frostie White; consulting engineer for nuclear applications Charles Heck; principal engineer for containment Necdet Kurul; Dan Pappone, the chief consulting engineer for plant performance; David Hinds, principal engineer for plant integration; Roseanne Harrington, manager of LOCA and containment analysis; and Guangjun Li, technical lead for LOCA and containment analysis.

In addition, we have Dr. Tom George, who is a consultant from Zachary Engineering. And with us today also are Bernard Gilligan and Jun Matsumoto from Hitachi America Ltd.

As questions arise, I may direct the questions to one or more of these individuals.

GEH is seeking NRC approval to apply an analysis method for evaluating the dry containment thermal hydraulic performance of the BWRX-300 small modular reactor, using the TRACG and GOTHIC computer codes.

The reason for this new method is because the BWRX-300 has a dry containment. Therefore, we are using a different code than we have used in the past for reactor and containment thermal hydraulic performance. Specifically, GOTHIC is now being used with inputs from TRACG to evaluate thermal hydraulic performance.

The scope of this topical report includes a method description and qualification, sensitivity studies, application of the method to the BWRX-300 for the events identified earlier or discussed earlier, and demonstration cases.

The analysis method used for the BWRX-300 containment thermal hydraulics performance demonstrates that the containment design complies with the following acceptance criteria, which are discussed in more detail in Section 4.0 of the Topical report.

The containment pressure boundary and penetrations are designed for the design pressure and temperature to be established for design basis accident in accordance with GDCs 2, 4, 16, 38, 41, 50, and 51.

The containment design pressure will be evaluated to bound the peak accident containment pressure resulting from the most limiting large break LOCA in accordance with GDCs 4, 16, 38, 41, 50, and 51.

The containment design features establish an essentially leak-tight barrier, and will be demonstrated to reduce containment pressure and temperature rapidly, maintaining them at acceptably low pressures following a LOCA, in accordance with GDCs 16, 38, and 50.

The containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from a LOCA in accordance with GDCs 16, 38, and 50.

I would like to emphasize that GEH is not seeking NRC approval for any exemptions from regulatory requirements.

I want to close out this section by restating that the methodology described in the topical report complies with all applicable regulatory requirements as written.

And GEH is not requesting NRC approval for exemptions from any regulatory requirements.

CHAIR MARCH-LEUBA: Lisa, this is Jose. Can you hear me?

MS. SCHICHLEIN: Yes. Yes, I can.

CHAIR MARCH-LEUBA: Yeah, because I opened my mouth before and it wasn't working.

In Slide 5 I noticed that we are asking for a 10 percent margin during the PSAR phase. That's right in the middle of the slide, the bullet No. 2. What do you mean by "during the PSAR phase"? It's not for the licensed facility?

MS. SCHICHLEIN: I'm sorry. Frostie, would you like to address that question?

MS. WHITE: I'm sorry if you can't hear me. There's a terrible thunderstorm going on right now, so.

CHAIR MARCH-LEUBA: We can hear you.

MS. WHITE: We, when we first wrote this section this was from when, as you said, two years ago. And in that part of it we were in our preliminary phrase. But this would be applicable throughout the life of the plant.

CHAIR MARCH-LEUBA: Okay. So, it's just a godfather sentence language.

MS. WHITE: Correct.

CHAIR MARCH-LEUBA: Thank you.

MR. WADKINS: Yes, this is George Wadkins. I did want to explain.

The 10 percent margin in the PSAR stage is actually a recommendation or a guideline within the SRP. It is an initial margin which is established in the old Part 50 process as the margin that you need to have at preliminary design stage, which equates to the PSAR stage for a construction permit application.

So, I just wanted to clearly state that. We do not necessarily have to maintain a 10 percent margin throughout the license of the plant. So, in the operating license application with the final safety analysis report that margin is not necessarily a requirement.

CHAIR MARCH-LEUBA: Okay. I will ask the

staff to remember in your presentation to see if you agree with this section. Okay? Thank you.

MS. SCHICHLEIN: Thank you.

Let's now move into the technical evaluation.

Section 1.3 of the topical report contains the acceptance criteria for the containment performance.

The acceptance criteria include:

Accident pressure and temperature less than the design pressure and temperature with an appropriate margin:

Containment pressure reduced to less than 50 percent of the peak accident pressure for the most limiting LOCA within 24 hours;

For LOCAs that do not produce the peak accident pressure, the containment pressure response after 24 hours is maintained below 50 percent of the peak pressure for the most limiting LOCA;

And for the containment atmos -- and the containment atmosphere remains sufficiently mixed such that deflagration or detonation does not occur inside containment.

The containment evaluation method presented in the topical report is an acceptable method to demonstrate that the above performance design criteria are met.

CHAIR MARCH-LEUBA: Lisa, can you describe in the open session what the containment atmosphere is during operation?

MS. SCHICHLEIN: I would like to pass that question off, please, to either Charlie Heck or Necdet Kurul.

Either Charlie or Necdet, if you could come off of mute and address that?

MR. HECK: Yeah, I think we can say here that -- Go ahead, Necdet.

MR. KURUL: No. Yeah, please go ahead.

MR. HECK: All right. This is, it's not proprietary, but this is inerted to containment.

CHAIR MARCH-LEUBA: Okay. Thank you. So, until you reach separation it's because there is so much inert gas, I will assume nitrogen, that there is no oxygen, there is no oxygen to deflagrate. Correct?

MR. HECK: Correct. Or very, very limited oxygen. Let's put it that way.

CHAIR MARCH-LEUBA: Yeah. But it's so diluted with the nitrogen that hydrogen and oxygen don't get to see each other.

Okay. Keep going.

MS. SCHICHLEIN: The reactor pressure vessel, including the isolation condenser, piping and reactor isolation valves, are described in the NRC approved topical report NEDC-33910P, "RPV Isolation and Overpressure Protection."

The conceptual containment design, penetrations, isolation valves, and Passive Containment Cooling System, are described in the NRC approved topical report NEDC-33911P, "Containment Performance."

There are several containment design features that are relevant to the containment evaluation methodology.

The containment is a dry enclosure, and near atmospheric pressure during normal operation.

It is also inerted with nitrogen during normal operation.

And there are no subcompartments containing large bore high energy lines.

And these subcompartments have sufficiently large openings so the boundaries of the subcompartments do not experience large pressure differentials from pipe breaks outside of the subcompartments.

Additionally, it should be noted that the containment design, pressure, and temperature are within the experience base of conventional BWRs.

Topical report Section 3 discusses the loss of coolant accident scenarios evaluated and limiting pipe breaks.

The limiting large pipe breaks evaluated are the main steam pipe and feedwater pipe with all large pipe breaks assumed to rapidly isolate at the reactor pressure vessel nozzle.

For small breaks, the limiting small break is an unisolated instrument line break.

Let's move into an overview of the containment evaluation model.

The containment evaluation method involves the use of two codes, TRACG and GOTHIC. TRACG is specifically used to evaluate the mass and energy release consistent with its use in evaluating mass and energy release for the ESBWR.

This method has been previously approved by the NRC as part of the ESBWR design certification.

GOTHIC is the code used to evaluate the containment thermal hydraulic response, and uses a new containment model developed for BWRX-300.

The GOTHIC code has been benchmarked to separate effect and integral tests. Benchmarking to the test data of a similar size containment is included in the topical report.

The containment response evaluation method for BWRX-300 uses both base and conservative cases.

For the conservative cases, key inputs, assumptions, and modeling parameters are simultaneously conservatively biased, which is the same approach used and approved for ESBWR.

Section 5.0 of the topical report provides details on the prior applications of TRACG.

The purpose of this section in the topical report is to demonstrate that the previous TRACG code qualification remains applicable to the BWRX-300.

The ESBWR TRACG LOCA method is applied to the BWRX-300 mass and energy release calculations to evaluate the large and small steam and feedwater pipe break base and conservative cases. The following bullets on this and the next slide outline the high-level content discussed in the topical report.

More specifically, the topical report provides details on the modeling biases in the Phenomena Identification and Ranking Table, or PIRT, as well as details on the nodalization and the initial conditions for the conservative cases.

To assist the NRC in their review of this methodology, GEH performed both base and conservative demonstration cases for the main steam and feedwater large breaks, as well as for the small steam and liquid pipe break cases.

GOTHIC then takes the output from the TRACG mass energy release calculations to evaluate the containment performance. Both base and conservative cases were evaluated.

As is done with the TRACG mass and energy release calculations, for the conservative cases the individual key inputs, assumptions, and modeling parameters are conservatively biased simultaneously, This is the same approach taken for the ESBWR containment method.

This next slide outlines the content provided in the topical report detailing the containment analysis method for the BWRX-300.

Section 6.0 of the topical report identifies the relevant inputs and phenomena relevant to the BWRX-300 containment response, and a selection of the models and correlations used to develop the base GOTHIC containment model.

It also includes the GOTHIC input model for the BWRX-300 containment, as well as the base cases and the results obtained from those base cases.

It includes a discussion of how nodalization impacts the calculated results.

There is also a discussion regarding which GOTHIC model uncertainties and biases are most important to the application of the GOTHIC model to the BWRX-300 containment evaluation methodology; the key model uncertainties and biases that are used in developing the conservative GOTHIC containment model; and the benchmark predictions of test data.

Section 6.0 of the topical report also includes the large steam line break and small steam or liquid break demonstration analyses showing the containment response for various break sizes and locations using the conservative GOTHIC containment model.

There is also a summary of the assumptions and inputs used in the GOTHIC conservative cases.

In conclusion, I would like to wrap up this presentation by restating that the methodology described in the licensing topical report complies with all applicable regulatory requirements as written.

GEH is not requesting NRC approval for exemptions from any regulatory requirements.

The method uses two computer codes, TRACG and GOTHIC.

TRACG uses the applicable parts of the approved topical report for the application of TRACG to ESBWR, which is incorporated in the approved ESBWR design certification.

And GOTHIC is a standard industry code used for thermal hydraulic containment evaluations in the nuclear industry.

Within the methodology, individual key inputs, assumptions, and modeling parameters are conservatively biased simultaneously, which is the same approach taken for the ESBWR containment method in the NRC-approved topical report Report NEDC-33083 P-A, Revision 1, "TRACG Application for ESBWR."

I'd now like to ask for any questions or comments for the GEH team.

CHAIR MARCH-LEUBA: Members, any comments for GEH on the open session?

I don't see anything. Thank you very much. Hold on.

Vesna, do you have any, any questions?

MEMBER DIMITRIJEVIC: No.

CHAIR MARCH-LEUBA: Thank you.

MR. SCHULTZ: I have a comment, Jose.

This is Steve Schultz. Lisa, a couple

just general questions.

In the staff presentation they indicate that we've seen a couple of reports here now on the containment evaluation, and also containment performance. There's another report from GEH coming in next on safety strategy.

Is there a nexus to the evaluations that we've seen here on containment and safety strategy, or is that something different? Just curiosity.

MS. SCHICHLEIN: I would like to have George Wadkins address that question, please.

George, if you're able to comment?

MR. WADKINS: Yes. This is George Wadkins.

There is only a limited amount of relationship between this and the safety strategy in that the safety strategy will define the functional safety functions that have to be met in the design as far as defense-in-depth. And, obviously, the containment function is one that we then design the plant to.

So, the safety strategy will define those defense lines necessary to meet the containment functional safety function requirement. And then also look at the postulated initiating events and sequences for AOOs and DBAs and severe accidents in order to establish when the containment function is required in the transient analysis.

So, that is the connection. The safety strategy establishes mainly functional requirements for containment. And then the transient analyses that are affected and required to be analyzed using the functional requirements of the containment.

MR. SCHULTZ: Thank you, George.

And my next question is related, well, it relates to what's coming up in the presentations by the staff with respect to the limitations and conditions that are proposed associated with their review.

And in those limitations and conditions the expectation is a requirement of either a particular or specified design and performance characteristic capabilities of the BWRX-300 design. Can perhaps get the issues addressed with some analysis demonstrations.

But are there any particular design characteristics or changes that you would anticipate as a result of a way to meet those limitations and conditions in a generic fashion so that licensees in the future don't have to address them? I presume with the BWRX-300 SMR that you want to have one design that you can apply across the customer base.

Is there any particular design features that you are considering changing to address the limitations and conditions?

MS. SCHICHLEIN: I'd like to see if Necdet, would you like to speak to that?

MR. KURUL: Okay. This is Necdet Kurul, GE-Hitachi.

Yes, there will be some designs to, design development to address some of the limitations and conditions. In terms of the isolation condenser radiolytic gas removal, there is a specific design being developed for that.

Also, isolation condenser return line design limitation and condition No. 2, that configuration is currently under development.

Limitation and condition No. 3, we do not believe that it is going to require any design change.

Limitation and condition No. 4, there will be some changes to passive containment cooling system. But we believe that is going to make the analysis even simpler than what is presented in this LTR, without changing any of the phenomena.

MR. SCHULTZ: Thank you for the response. I wanted to see that on the public record.

Thank you.

CHAIR MARCH-LEUBA: And just for clarification on public record, those will be part of the final design. We are only reviewing now how the methodologies work with or without them. Is that correct? Basically, we're not assuming they have been implemented, and we don't need to because we're reviewing the methodology of how we will arise them in the system.

MS. SCHICHLEIN: Correct.

MR. KURUL: That's correct.

CHAIR MARCH-LEUBA: Thank you.

Any more questions, including Vesna?

(No response.)

CHAIR MARCH-LEUBA: So, this ends the open portion of the GE presentation. And we're going to transfer to the staff. Now we will have the staff presentation.

So, we're going to see the open session for the staff.

So, I want to mention for the people that are not present that it's a big milestone today. For the first time in two years we have actual in-person presenters in this room. We're very happy to see you. And it is a new configuration. So, keep your microphones on red unless you are talking. And when you want to talk, put them on green.

So, and you are going to be sharing the slides?

MR. SHEA: Yeah. I was just going to ask permission, this is Jim Shea, I was going to ask permission to share my slides, my screen.

Understand.

CHAIR MARCH-LEUBA: You're the presenter; right?

MR. SHEA: Yes.

MR. DUDEK: Jim, Subcommittee Chairman March-Leuba, I think Scott has some opening remarks as the responsible branch chief, if we do have a minute?

CHAIR MARCH-LEUBA: We always have time for that. Please go ahead.

MR. KREPEL: Can you hear me?

CHAIR MARCH-LEUBA: Yes, we can.

MR. KREPEL: Okay, hello. Hi. This is Scott Krepel. I'm actually speaking through a sign language interpreter.

I am the Branch Chief for the Lead Technical Review Branch, Nuclear Systems Performance Branch. This is a complex review with staff from four different branches, as well as a novel coordinator with the Canadian Regulatory Agency.

Some of you may recall that before I became a branch chief I was a topical report reviewer. I can say that I've seen comparable reviews take two to three years, or even longer. The staff did a good job of completing its review in just a little more than one year, despite some unexpected challenges.

I would like to commend the GEH staff for being very responsive to our questions and requests, which was crucial to our ability to support this schedule.

Thank you, everyone.

MEMBER REMPE: Well, I have a question.

I'd like to hear a little bit more about what happened with the coordination with the Canadian review. And I don't know if Scott or one of you could answer what impact it had and how it progressed?

Be sure and state your name.

MR. LU: Shanlai Lu from staff. And I'm Acting Branch Chief for Branch 7.

So, we do have the agreement signed off between two agencies regarding the BWRX-300. And my understanding is the Canadian has, you know, approach and is in the process to review six BWRX-300 units.

So, their CNSC staff is working closely

with the NRC staff through the entire review process. And we have, we shared all the proprietary information between the GEH and CNSC. And so, actually they are on this, too.

MEMBER REMPE: But, again, I'd like more detail. Is it healthy? Is it taking longer? I assume it will make it easier in the long run. But are they, how do you do this? Will you divide it, say, okay, you take?

MR. LU: Yes. You can see Mike Dudek is the Licensing Branch, can talk to that.

CHAIR MARCH-LEUBA: Yeah. Mike wants to talk to that.

MR. DUDEK: So, it's a very good question. This was a joint effort in every way, shape, and form. They actually, we actually split, swapped some staff members. The NRC sent a staff member to go up, virtually to go up to Canada and learn about their process, procedures, and integrate with their staff during their portions of the review.

And they had a staff member come down and attend just about every working group session, and every technical discussion, and every audit discussion that we had on this.

So, with that being said, certain parts

weren't assigned to certain organizations; it was a collaborative effort. You know, we, instead of having, you know, a technical staff member sitting down at a desk and doing the technical review solo, you know, they would bring into a group atmosphere and have a group discussion on some of those items. And then they would jointly write it in the SharePoint file, then have, and have that dialog and have those discussions as they went.

So, it's a true joint effort in every way, shape, and form. And we actually developed a joint report that we're very, very close to bringing to closure. I spoke to division management today and, hopefully, going to ops management next week. And hopefully out sometime in early August in which we developed a lot of these thoughts to help explain how we conducted the review, the things we found, some mutual learnings, and some good taskings that we could take back and use during both of our reviews when we have the applications come to pass.

Does that answer your question, Chairman Rempe?

MEMBER REMPE: Yes. I just am curious. I apologize, but I can't help ask one more question.

Did you see that they had certain aspects they were focused more on than the U.S. or vice versa? Or is that too much to ask and I ought to just be patient and wait for the report?

MR. DUDEK: So, a lot of that's outlined in the report. They definitely have a different perspective. I believe that the NRC's reviews are more focused on the codes and running those independent analyses. And I think that we learned a lot about each other's regulatory structures and how we do the reviews.

I think that can be summarized as the majority of the learnings. So, very beneficial for both parties to get that perspective.

MEMBER REMPE: Well, that's great. Send the report when it's available to my staff and we can distribute it to us. But I appreciate the time to divert on this topic.

Thank you.

MR. DUDEK: Absolutely. And we hope to have that report finalized and sent to you before the full committee meeting. That's our goal.

MEMBER REMPE: Great.

CHAIR MARCH-LEUBA: Yes. We are going to see within a week-and-a-half. Okay.

MR. SHEA: Just to have a slight add to

that. This is Jim Shea again, the project manager for the BWRX-300 for the NRC staff.

Just that they participated in the audit process. The audit went almost a full year. It was every week we had a meeting. And number of issues, approximately 45 issues that were resolved. Some ended up in RAI space. But it was very extensive, and the Canadians participated in that process.

Good afternoon. My name is Jim Shea. Like I said, Office of Nuclear Reactor Regulation, the Division of New and Renewed License, DNRL. I'm the project manager for the GE BWRX-300 design licensing topical report, and the lead project manager for the, overall for the GEH BWRX-300.

I want to thank the ACRS full committee for this time to allow the staff to present its review of this licensing topical report as reflected in the staff safety evaluation report which was submitted publicly March 9th, 2020 in ADAMS.

The staff review focused on the approval of the GEH BWRX-300 containment evaluation method. You may see an acronym used: CEM. You'll see that's Containment Evaluation Method. For the evaluation of the design basis accidents as stated in the safety evaluation report.

The NRC staff will evaluate the compliance of the final BWRX-300 SMR design, including the containment evaluation method, during future licensing activities in accordance with 10 C.F.R. Part 50 or 10 C.F.R. Part 52.

The LTR was submitted to NRC on September 25th, 2020. Revision 1 was supplemented with RAI responses and was submitted to the NRC on November 19th. And additional Revision 2 was submitted December 17th, 2021

I just want to go quickly as the lead TM to look at our status in the big picture for pre-application activities.

You can see that during the presentation the DEH and the NRC we referred back to some of these LTRs that had been approved. You could almost look at the approved containment performance as a precursor for this LTR. They kind of fit together, the requirements that were outlined and that became the performance LTR are what are being addressed in this LTR as far as the DBAs and criteria.

Two LTRs are currently under review, the two that we had viewed today. And there's three more expected this year. Sometime in June a safety strategy methods that we talked about earlier, and a severe accident management source term methodology later. Also an instrumentation and controls architecture will be coming out.

CHAIR MARCH-LEUBA: Jim, I always start my questions with can you describe in open session what the safety strategy LTR is about, high level? It was mentioned earlier. I know what it is but I don't know if it's proprietary.

MR. SHEA: Yeah, I don't believe it's proprietary. I think GEH already kind of answered it.

They're looking at their safety strategy method. It's taking the IAEA methodology for looking at safety systems, really classification of safety systems, and using that hierarchy, and seeing how this reactor will fit into that hierarchy. Even though they're still going to meet all of our, you know, our regulations. But they're just taking that because that, you know, for their, you know, the Canadians are using that architecture for it.

CHAIR MARCH-LEUBA: My understanding, and I haven't seen the topical, is they're going to align the IAEA requirements with the NRC requirements. Whatever is higher --

MR. DUDEK: Chairman March-Leuba, yes, I would refer to maybe, perhaps, GEH to answer the specifics behind their intent on these topical reports. I would hate to speculate, since we haven't seen, you know, in-depth details about what is included in these, the NRC staff.

CHAIR MARCH-LEUBA: We will ask them in the closed session.

MR. SHEA: Thanks, Mike.

I can refer to you there is some of that discussed in the LTR that's already -- that has been approved on the activity control. There is some discussion about that criteria in the hierarchy that's kind of confusing folks. But that's --

MR. BROWN: This is Charlie Brown. The instrumentation control architecture, and don't take this, my comments the wrong way, I'm saying this talking. We first reviewed this, or I first reviewed this back in 2009 for ESBWR. All right. And it was difficult. The architecture is not well defined, very little detail relative to what we've done in the rest going up to APR-1000, NuScale, APR-1400.

But the last two went very, very smoothly because it addressed what the committee's been looking for: good architecture diagrams, good explanations of control of access, independence, all the stuff that goes into the fundamentals.

Is this just going to be a repeat of the ESBWR design? Because it took almost a year-and-a-half or two to finally get any change of I&C or manager.

MR. SHEA: I think, I think that what Mike said is accurate. We don't really know at this stage. They haven't been submitted.

But I think GEH is listening. So, I think they probably took your comment.

MR. WADKINS: Yes. This is George Wadkins from GEH.

I understand your comment. We are, we are going to follow on with the more later applications, including the NuScale application approach. So, we believe that we will be able to hit the mark based on the more current approval of I&C architecture.

MR. BROWN: Okay. That will be excellent. Thank you. Makes it very, very easy when that architecture is well defined, including the definition of one-way hardware-based non-software configured communication devices in a network, et cetera.

So, does that --

MR. LU: This is Shanlai Lu. Thanks for the comments.

I think this, for the purpose of today's presentation. It's a part of this topical report is the continuing methodology. Whether we hear that one --

MR. BROWN: I'm just trying to make sure.

MR. LU: Yes.

MR. BROWN: So, early understanding of what we might be expecting.

MR. LU: Got it.

MR. BROWN: At least what the committee's going to be expecting to be able to do.

MR. LU: Got it. We got it.

MR. BROWN: That would be helpful instead of waiting for six months to say that's not --

MR. LU: I agree. We all agree. We all agree with that.

MR. BROWN: All right. Thank you. I will stop.

CHAIR MARCH-LEUBA: Shanlai, you might start thinking that maybe we wasted 10 minutes on this. But this is very useful for the membership, good to know what's coming and that there are labor --

MR. LU: Yes.

CHAIR MARCH-LEUBA: -- and we have a lot of work. Thank you. Thank you for that.

MR. SHEA: Okay, thank you. And we will

get back to the task at hand.

And the next slide we have the NRC staff review team. And I'll turn it over to Syed Haider, who is the lead reviewer for this, for this LTR.

MR. HAIDER: Thank you, Jim.

Good afternoon to everyone. My name is Syed Haider. I'm the lead reviewer on the GEH licensing topical report LTR NEDC-33922P on BWRX-300 containment evaluation method, which is mainly related to modeling the containment pressure and temperature response that were reserved from the limiting design base events involving mass energy leaks from the reactor pressure vessel into the containment for the BWRX-300 SMR design.

My Nuclear Systems Performance Branch at NRR had the lead for the review under Branch Chief Scott Krepel; Jim Shea, the NRR New Reactor Licensing Branch GPM.

Carl Thurston and Shanlai Lu from NRR's Nuclear Methods, Systems & New Reactors Branch were the reviewers of the LTR part related to the TRACG modeling and mass energy release calculation methodology, that is covered under Section 5.0 of the safety regulation report.

I, myself, mainly reviewed the GOTHIC

modeling and containment pressure and temperature response calculation methodology that's covered under Section 6.0 of the SER.

Chang Li from Containment and Plant Systems Branch reviewed combustible gas, gases inside the containment. Throughout the review the NRR staff was supported by the research staff that includes Peter Lien, Joe Staudenmeier, and Andrew Ireland from the Code and Reactor Analysis Branch, and Shawn Campbell from the Fuel & Source Term Code Development Branch.

The staff were responsible for the development of the TRACE and confirmatory models that were used to analyze and relegate the TRACG and GOTHIC model reserves. And terms presented in the LTR we can provide those details during the closed session.

Next slide, please.

This slide shows the outline of the current open staff presentation during which we will avoid discussing any proprietary information. That will be covered during the closed session later.

First, the staff will give an overview of the intended purpose and scope of the BWRX-300 LTR. It's worth mentioning here that even though the LTR title is BWRX-300 containment evaluation method, the

LTR covers the methodology and modeling details for both the BWRX containment as well as the operating site.

First I'll go with the BWRX-300 containment design background that drives the code of the presented methodology.

We will review the components of the BWRX-300 design as they relate to the TRACG and GOTHIC code based RTV and containment analysis methodologies with the eventual objective to predict the containment thermal hydraulic response with sufficient concept data.

In this regard, we will explain the regulatory requirements and BWRX-300 acceptance criteria for the containment response underlying the LTR and how they relate to the present safety evaluation for the design basis accidents.

Then I will give a summary of the BWRX-300 containment evaluation method, demonstration and analyses that I presented in the LTR.

After these introductory slides, my NRR colleague Carl Thurston will present a few slides on the TRACG code based mass energy release calculation methodology review. After that, I'll present a few slides on our GOTHIC code based containment response calculation methodology.

The detailed technical research for the TRACG is mass energy methodology, and GOTHIC is containment response methodology, are proprietary and will be presented during the closed session.

Again, Carl and I will describe the full conditions and limitations on the LTR as they emerged during the staff review of the LTR.

Then we will finish the presentation of the staff reviews.

Next slide, please.

This slide describes the scope of the BWRX-300 containment evolution methodology LTR as reviewed by the staff. Basically GE-Hitachi has submitted this LTR to obtain the NRC staff's approval of the overall BWRX-300 containment peak pressure and temperature analysis methodology, as presented in the LTR.

The NRC regulations and acceptance criteria that deal with the BWRX-300 containment thermal hydraulics performance are listed in a separate GE-Hitachi licensing topical report on BWRX-300 containment performance. That was presented to ACRS last year.

In the future, the approved methodology

will be used to perform the BWRX-300 containment safety analysis to support either a licensing application for a construction permit and operating license under 10 C.F.R. Part 50, or a design certification and combined license application under 10 C.F.R. Part 52.

Next slide, please.

So, I now will summarize the state-of-the-art of the BWRX-300 containment design and its relation with the current staff review and the methodology. And this slide just presents the salient design features of the BWRX-300 containment that are relevant to the containment safety analysis methodology for the design basis accidents.

These design features drive several initial and conditions in the containment GOTHIC model. So, basically BWRX-300 has a dry containment that's inerted with nitrogen during normal operation.

It does not have a suppression pool inside the containment like ESBWR had.

This information is relevant to initial containment pressurization and the post-accident mixing of steam and radiolytic gases inside the containment.

A key design feature of the BWRX-300

design is the early closure of its RPV isolation valve that limits the mass and energy release into the containment during the large break LOCA which dictates the peak containment pressure.

However, the RPV isolation valve does not close for small breaks that remain unisolated resulting in a continuous break flow for the small break LOCA analysis period.

And the BWRX-300 design employs a passive containment cooling system, or PCCS, that's used to remove heat from the containment reactor cavity pool that's located above the containment. PCCS plays an important role in the long-term containment pressure reduction and mitigation.

In this regard, the LTR demonstration analyses were performed with the specific PCCS design unit described in the LTR. The reactor cavity pool that's used by the PCCS for containment located above the containment.

So --

MEMBER HALNON: I have a quick question. This is Greg.

Small break LOCA, as I mentioned, there was an instrument line.

MEMBER HALNON: -- the limiting break?

MR. HAIDER: Limiting break for small break LOCA.

MEMBER HALNON: It is a limiting break?

MR. HAIDER: Yeah.

MEMBER HALNON: You're talking about half inch, quarter inch? What's the instrument line size?

(Simultaneous speaking.)

MEMBER HALNON: Sorry.

CHAIR MARCH-LEUBA: Yeah, let me remind everybody of the new rule. The microphones are really good for the people that are outside of the room. But they don't amplify anything inside of the room. So please speak. When you speak into the -- when you talk into speaker -- microphone, people outside the room hears you very well but we don't. So, like, we're not hearing you.

MEMBER HALNON: Soft spoken. Got it.

MR. HAIDER: So the staff has performed this review with a clear understanding that the BWRX-300 design is not final yet. And the analysis presented in the LTR are demonstration cases. The objective of the present review is to evaluate the applicability of the methodology, the BWRX-300 design, and access the degree of conservatism.

This has required the staff to focus on

the uncertainty and the methodology and predicted trends without getting unduly exact numbers. The staff has the understanding that the input and output numbers may vary in the final design. That would be submitted in the licensing stage.

So the overriding question the staff resolved was how to approve the evaluation methodology even if the BWRX-300 or similar design has not been evaluated. Next slide, please. This slide highlights the four acceptance criteria for containment response used in the methodology. These acceptance criteria are identified in the LTR for the BWRX-300 design safety analysis and are essentially driven by several NRC regulation.

This slide also captures two key regulatory requirements for the containment safety analyses that deliver the short term peak containment pressure and temperature and the long term pressure and temperature response of the containment. One key regulatory requirement that is rooted in GDC 16 and 50 of 10 CFR, R50, Appendix C is to ensure that the peak containment pressure and maximum wall temperature calculated for the limiting mass-energy release design basis accidents are bounded by the BWRX-300 design, pressure, and temperature of its sufficient margin.

I believe the question that was raised during the presentation by GEH was about the 10 percent. The 10 percent margin is an interpretation of SRP Section 6.2.1.1.A. All of them were sufficient to use in GDC 50.

So GDC 50 requires sufficient (audio interference). SRP Section 6.2.1 interprets it as 10 percent. So it's a guidance. It's not a regulation. It's not a requirement.

CHAIR MARCH-LEUBA: The 10 percent really didn't confuse me as much. It could be 8, could be 12. What confused me is that it applies to the preliminary SAR and doesn't apply afterwards. That's what I thought George Wadkins said and certainly what the slide implied. So the sufficient margin would apply to the final design.

MR. HAIDER: That's right.

CHAIR MARCH-LEUBA: And we will have to agree whether 10 percent is sufficient or not.

MR. HAIDER: Correct. That's right.

CHAIR MARCH-LEUBA: All right. So when we have a final design, you will come back here and tell us what sufficient is. Thank you.

MR. HAIDER: Yeah, one more clarification that I would like to make here is that the word, margin, exactly is not relevant to this review. The correct word is conservatism. The staff focused on the conservatism. The margin when it comes up in the design is final. That would be the difference between the final design attributes and the limiting conservative results.

CHAIR MARCH-LEUBA: Yeah, I can see your point.

MR. HAIDER: So the second key regulatory requirement is driven by GDC 38 of 10 CFR Part 50 Appendix A for containment heat removal that is interpreted by the standard review plan as to ensure that the containment pressure is reduced to less than 50 percent of the peak pressure for the most limiting LOCA within 24 hours. So that is for a long term pressure response. GDC 38 also requires that the containment pressures 24 hours after LOCA is maintained below 50 percent of the peak pressure for the most limiting LOCA.

So not only the depressurization should get the containment to less than 50 percent in 24 hours, the pressure should as remain low, that the mitigation can be interpreted. So essentially the first three acceptance criteria, GDC 50, 16, and 38 that we have described, collectively ensure that the containment structure can accommodate the pressure and temperature condition exerting from any mass and energy release from the RPV without exceeding the design decay rate and it's sufficient margin. The full acceptance criterion requires that the containment atmosphere remains sufficient mixed so that the deflagration or degradation does not operate inside the containment.

CHAIR MARCH-LEUBA: Using GOTHIC, can we estimate mixing? Or is that based on first principles? I don't see how GOTHIC predict that the atmosphere remains sufficient mixed, do you?

MR. HAIDER: They have covered benchmarking that I was going to talk later, CVTR. And that has certification. And the benchmark showed benchmarking of GOTHIC code against the certification for the CVTR.

CHAIR MARCH-LEUBA: Also, you're saying that GOTHIC is actually qualified to identify certification?

MR. HAIDER: That is my understanding. And that's also the information they have included. Next slide, please. So the LTR presents the methodology demonstration in L&Cs and reserved for evaluating the BWRX-300 containment and hydraulic performance against the acceptance criteria for the limiting design basis accident. In this backdrop, the LTR presented alerts for large-break loss of coolant accident and small-break loss of coolant accident inside the containment.

The containment design basis analysis presented in the LTR include both liquid and steam breaks. This shows that the four acceptance criteria were satisfied for the demonstration of cases. However, the applicant would have to demonstrate meeting the acceptance criteria for the final BWRX-300 design at the licensing stage while meeting the four conditions and limitations the staff has imposed on the LTR at the start of this review.

As mentioned earlier, the methodology uses TRACG code to calculate mass-energy release from the RPV into the containment. And GOTHIC code is used to calculating exerting containment response. The LTR provides the TRACG later detailed in Section 5 and GOTHIC detailed in Section 6 of the LTR. And now I will ask my NRR colleague, Carl Thurston, to present the public summary of the TRACG modeling of the mass-energy release for BWRX-300 containment.

MR. THURSTON: So my name is Carl Thurston, and I am the primary reviewer for the TRACG code calculation of M&E releases. The TRACG code has a long history of review by NRC staff to disable previous topical reports submitted, including ESBWR, ESBWR, and the existing fleet of BWR 2 to 6 reactors. For review of the code, the staff focused its review on code changes made since ESBWR.

And staff basically determined that there were no significant changes since the analysis was completed for ESBWR. The RPV model and internal components were essentially scaled from ESBWR. The methodology uses a decouple method so that the containment is maintained at atmospheric pressure throughout the full event, whether it's a small break or a shorter time -- or excuse me, large break for a shorter break or a small break for up to 72 hours. The containment --

(Simultaneous speaking.)

CHAIR MARCH-LEUBA: So there is no feedback from GOTHIC back into TRACG?

MR. THURSTON: That's correct.

CHAIR MARCH-LEUBA: And that is conservative because we estimate a larger flow --

MR. THURSTON: That's correct.

CHAIR MARCH-LEUBA: -- out of the vessel?

MR. THURSTON: That's correct. The

methodology relies heavily upon previous TRACG topical reports and their compatibility with BWRX-300, particularly for the ESBWR qualification. So the ESBWR qualifications extended to BWRX-300 such that ESBWR PIRT and model biases apply directly to the RPV and internals.

In addition, the BWR/2 through 6 methods were invoked because there were some events where core uncover results and you need a more detailed model to resolve the phenomenon. The isolation condensers have a significantly more important safety function than was used for ESBWR. And so it's modeled in considerably more detail for the BWRX-300. Staff determined that the modeling was adequate for M&E release calculations with consideration of the limits and conditions being applied. Next slide.

CHAIR MARCH-LEUBA: Oh, of particular concern, everybody will tell you -- you will tell us. Is the presence of non-condensable isolation condenser? How will this TRACG model non-condensable gases?

MR. THURSTON: My understanding is that TRACG can model non-condensable gases within the code. And we'll get into this more in our proprietary session.

CHAIR MARCH-LEUBA: Okay. We'll wait --

(Simultaneous speaking.)

MR. THURSTON: That was the reasons why they did include it. So the next slide focuses on unique features. And I think this has been covered and fairly detailed already. But the larger breaks are all isolated by valves that are mounted on the nozzles of the RPV that close quickly so that any release from a larger break is limited.

There's no suppression pool. And the isolation condensers are the primary. They are the only mechanism of decay heat removal for this plant. RPV isolation valves limits break flow and mass release indicated.

But the small breaks are un-isolated and continue to blowdown for 72 hours. For this analysis, one of the trains of ICS was considered inoperative due to single failure. The determined that conservative inputs were used for initial power, power history, scram, choke flow model, and the fact that they used atmospheric boundary condition. And in addition, the operating conditions also were conservative. Next slide.

So there were two key issues that the staff found and related to the TRACG calculation of mass and energy release. The first one is related to accumulation of radiolytic gases and the isolation condenser. The staff looked at the methodology that was used by GEH and did not feel the gases were appropriately modeled.

The staff did confirmatory analysis that we will show in a closed session. So those gases that can be generated in the reactor can transport to the isolation condensers. And if they get there, they will heat transfer and the isolation condensers are not able to perform as rated.

CHAIR MARCH-LEUBA: And the most direct pass is a return line, not the forward line?

MR. THURSTON: So both passes are potentially allowing radiolytic gases to get to the IC, whether it goes to the steam line or through the return line.

CHAIR MARCH-LEUBA: And the solution with this recommendation, they will talk more detail in the closed session. It's to either license better it better or fix it.

MR. THURSTON: Yeah, so we will get into that.

CHAIR MARCH-LEUBA: Yeah, we'll wait.

MR. THURSTON: So as we indicated, the

non-condensable gases will build up in a lower drum. And that degrades the heat transfers. It can significantly degrade heat transfer in isolation condensers.

The second issue that the staff found and write an RAI about is related to the design of the return line. So for the ESBWR, the return line was discharged into the downcomer. And for this plant design, if I'm going too far, GEH will correct me, it discharges into the chimney region which is a slightly higher pressure.

So it increases the likelihood that steam can back flow into the return line and again make it to the isolation condenser and in effect heat transfer in the isolation condenser. So that's the first and second limited conditions. The next slide --

(Simultaneous speaking.)

CHAIR MARCH-LEUBA: Is the concern that steam will go through the back flow through the return line or non-condenser?

MR. THURSTON: Steam and non-condensers will be in the steam. And when it condenses --

(Simultaneous speaking.)

CHAIR MARCH-LEUBA: But the following will be the -- I mean, if more steam makes it to the ICS, it's working better. ICS works like condensing steam. The more steam you put up there, the more liquid you're sending down.

MR. THURSTON: That's right.

CHAIR MARCH-LEUBA: The problem will be the non-condensable --

(Simultaneous speaking.)

MR. HAIDER: Hi, this is Syed Haider back to cover the GOTHIC and containment slide of the review. This slide presents an overview of the GOTHIC code as reviewed by the staff of the BWRX-300 containment response and the LTR application for the NRC approval. GOTHIC code is an established computer code that has been widely used in the industry for containment response analysis.

This is a continuously improved code that is compliant with the pertinent NRC regulation, 10 CFR Part 50, Appendix B. GEH has used the latest GOTHIC version 8.3 of the BWRX-300 analysis demonstrated in the LTR. During the order, GEH clarified that they followed the relevant analysis process should there be a need to use a different GOTHIC version for the final design of the licensing stage.

As far as the GOTHIC based applications in the past are concerned, GOTHIC code has been previously approved by NRC for containment response analyses. The staff also reviewed BWRX-300 containment PIRT method phenomenology and found it covered by the available GOTHIC functionalities. The staff also reviewed the GOTHIC benchmarking against the CVTR test data that the LTR presented for the BWRX-300 containment methodology qualification.

The information provided in the LTR and RAI responses demonstrated that GOTHIC is qualified for the thermal stratification and 3D effects. These features are important for the nodalization-based containment design approach of distributed PCCS units. Next slide, please. This slide captures the key features of the GOTHIC model development under containment response calculation methodology and the LTR.

The evaluation method for the BWRX-300 containment response for design basis accident has been developed following the applicability of Reg Guide 1.203 that is used to establish an acceptability evaluation method and analyze transient nuclear power plant responses during the postulated design basis accidents. Reg Guide describes a multi-step process for developing evaluation models for assessing well defined figures of merit. The BWRX-300 containment evaluation methodologies based on decoupled calculation between TRACG and GOTHIC where TRACG RPV model calculate the mass-energy release into the containment assuming it remains the same initial atmospheric pressure throughout the accident duration. That means that the TRACG RPV model mass-energy release calculations do not account for the increasing containment backpressure.

We're using the decoupled approach. The standalone GOTHIC containment model independently uses the mass energy release calculated by TRACG as a containment boundary condition to calculate the containment pressure and temperature response. The staff agrees that this ensures a conservative mass-energy release into the containment with respect to containment pressure and temperature response. As described in the LTR, the containment evaluation method is based on four component GOTHIC model that include a nodalized mean component of containment, a nodalized dome, a nodalized PCCS, and a lumped reactor cavity pool in aerospace.

A Conservative Diffusion Layer Model, DLM, is used and justified for modeling condensation heat transfer on the containment shell and PCCS surface. The option for condensation built into GOTHIC to calculate the condensation rate is based on a mechanistic model of the underlying phenomena which is based on heat and mass transfer analogy. A nodalization-based modeling of fluids accounts for the thermal stratification inside the containment. Next slide, please.

This slide summarizes the staff review of the BWRX-300 GOTHIC containment response methodology. More information will be provided about these events during the closed session. The staff reviewed the applicability of the important physical phenomena identified by the GOTHIC PIRT table for the BWRX-300 design and found them appropriate.

The applicants submitted GOTHIC decks based on conservative cases for both big and small break LOCAs, we reviewed the GOTHIC input model and used the submitted GOTHIC decks to gather information to develop the TRACE and MELCOR decks for confirmatory analyses. Key modeling and uncertainties and conservative biases were reviewed to establish the overall GOTHIC model conservatism. We'll review and we'll provide more information about the nodalization sensitivity studies for the containment and PCCS during the closed session. We also reviewed the LTR benchmarking of the GOTHIC code against the CVTR test data and the justification for bounding the containment, condensation, and natural convection heat transfer correlations by using the coping (phonetic) test data.

The staff also made sure that the BWRX-300 containment response analyses do account for appropriate break location and break flow direction sensitivities. The staff also reviewed the PCCS capacity to reduce and mitigate the containment pressure in the long-term, identifying a small un-isolated liquid break as a potential limiting small break LOCA. Containment mixing was also reviewed with respect to combustible gasses accumulation inside the containment. The staff also performed confirmatory analyses to develop necessary insights during the review.

CHAIR MARCH-LEUBA: I've seen the proprietary version of this. And you kind of put the confirmatory analyses at the end as bullet number 10. And it was a pretty significant effort. I wanted to emphasize in the open session that there was a significant effort by the staff to do confirmatories. And I confirmed that if it works.

MEMBER REMPE: And I don't know what you can say in the open session. But can you saw the tools that you used? It's in the SE. Is that not allowed to say what codes or methods you used --

(Simultaneous speaking.)

MR. HAIDER: You mean the confirmatory --

MEMBER REMPE: Analysis was done with?

MR. HAIDER: Was with TRACE and MELCOR codes.

MEMBER REMPE: Yeah, I just would like to have that.

MR. HAIDER: Yeah, TRACE and MELCOR codes were used to develop the confirmatory model. Yeah, it was a significant part. And that complimented our staff because significant insights came from the confirmatory analysis that the research conducted throughout.

MEMBER REMPE: Thank you. I just would like to have it in the open.

CHAIR MARCH-LEUBA: Yeah, while we're on the record -- you found a key point. It's not the confirmatory analysis per se. It's the inside of the staff review and the design that you obtain by having to do the confirmatory. That provides significant value. That insight is what's invaluable.

MR. HAIDER: That's true. That's very true. Thanks for highlighting that. Next slide, please. This slide gives a summary of the key technical issues that the staff encountered on the containment side of the review. We will present the needed technical details of the issues resolution and the supporting confirmatory analysis (audio interference) are during the closed session.

The already submitted Rev 0 of the LTR and not performed any break location or break flow direction studies -- sensitivity study. The staff will elaborate on its concerns about the limiting break location and flow orientation during the closed session. The staff also noticed that limiting a small break was a steam break LOCA in the LTR who had no containment analysis was performed for this small liquid break LOCA.

So RAI is where it should and their response showed sensitivity to the break location and break flow direction. We've modified the limiting large break location and break flow orientation. It responds also -- showed that a liquid rather than a steam break while the limiting small break LOCA with respect to peak containment pressure.

The staff had noticed that even though the applicant had presented a containment and nodalization study for large break LOCA but with four different nodalization schemes. Non-light sensitivity study was performed for the small break in the original submittal. So an RAI was issued and a small break LOCA nodalization study revealed that even though the peak containment pressure was sensitive to the nodalization of the scheme, it was always smaller than the limiting peak containment pressure that occurs during the large break LOCA for the containment design and used in the demonstration analyses.

It was also observed that the sensitivity to small break LOCA nodalization diminished in the long-term response. However, the deck point, the applicant try to go around the containment pressure response by using the limiting RPV pressure response in a quasi-best estimate fashion. Those details will be provided in the closed session.

However, using the limiting RPV and containment pressure for SBLOCA for the long term containment response evaluation raises a distinct possibility of break flow reversal on containment break to the RPV with non-condensable gases on the containment entering the RPV and degrading the isolation condenser. This has led to the development of limitation and condition number 3 that essentially requires that the BWRX-300 final design has to be demonstrated for no safety significant break flow reversal. This will be further discussed during the closed session.

CHAIR MARCH-LEUBA: Let me stop you right there because this is key. Because if you generate radiolytic gases inside the vessel, hydrogen, and you can recombine it with oxygen and get rid of it. But if you input a lot of nitrogen into the vessel, there is nothing you can do short of bending it.

So when we have a final design, we will expect you to look at this design and make sure no reverse flow happens with sufficient confidence. I'm certain (audio interference) know everything. But allowing a lot of nitrogen into the vessel is bad for this design. You know that.

MR. HAIDER: Yes, that's correct. That was the prime mover on limitation and condition number 3. So this will be -- of the condensation and natural convection heat transfer biases used in the model, and significant differences between the GOTHIC of predicted dome and PCCS heat transfers in the confirmatory analysis.

The applicant deep dived into the GOTHIC decks and also identified a major error in the PCCS condensation heat transfer modeling. This was a significant correction to a large number of plots in the LTR that left to the revision of an LTR revision -- one LTR revision. We'll provide further details of the error and the resolution in the closed session.

The staff also showed an RAI on the PCCS modeling and performed a PCCS confirmatory study to validate the PCCS sensitivity study presented in the LTR. More detail will be provide about these staff concerns and their disposition. In the RAI, the applicant did mention that even though they have presented the demonstration cases with a specific PCCS configuration.

And they may use a different PCCS configuration for the final design. In this regard, the staff has introduced limitation and condition number 4 for the applicability to the final PCCS design for licensing basis. Now I'll ask Carl Thurston to describe the two RPV-specific limitations and conditions that we attached.

MR. THURSTON: Thank you, Syed. So now we'll basically just review those two limit and condition. I think we described them I think in fairly good detail already. If there's any questions, I think we can address those.

But the first one is related to the

accident, if the normal steam going into the isolation condenser can accumulate, it degrades the effectiveness of the isolation condenser. And again, because the isolation condenser is the primary path of decay heat removal, it has to function correctly and at the rate it conditions. The next limit and conditions are related to the return line.

And again, the main emphasis is to prevent the flow of radiolytic acids back to the isolation condenser and not disrupting the flow. Also there's reverse flow, it can affect the amount of flow that's going to the -- back through the lines. So the line has to flow efficiently to keep the whole cycle working. So that's the reason for limit and condition number 2. So I'll turn it back over to you, Syed.

CHAIR MARCH-LEUBA: Before you move to the next, the 72-hour period in my opinion is a leftover from active large lightwater plants where LOCA and ECCS failure was dependent on loss of power. And we decided 72 hours was sufficient long enough to restore it. Now here we have passive plants. Does the 72-hour still make sense or do we need to make a week, a month?

MR. THURSTON: So we are still using the 72-hour as an analysis. Maybe I can turn it over to

Shanlai Lu. It's been our guidance and our practice. I don't know if it's codified how we apply it directly. So Shanlai can clarify that question.

MR. HAIDER: Can I offer my comment? Syed here. I engaged GEH on this specific issue of 72 hours and the applicability of the methodology through an RAI and also through audit discussions. Their outlook pretty much was that this methodology is driven by, in part, phenomena identification table.

So their position was that as long the phenomenology remains applicable and doesn't change for any reason, they believe that the methodology should or could be extended beyond 72 hours. So the acceptance criterion is the applicability of the phenomenon rather than the 72 hours deadline.

CHAIR MARCH-LEUBA: I understand why you're saying that. Today we're talking about the SER on this topical report which is a methodology. My question -- and I apologize. I maybe should have ask it today or maybe I should ask it two years from now.

If we survive -- if the X-300 design survives for 72 hours and we know it's on trend that it's going to fail in two hours later or two weeks later, would that be accepted? And this is a bridge we'll have to cross two years from now when we review the X-300. As of today, we're looking at the methodology -- and what you say is correct, the methodology -- is applicable here.

That's what we're concluding today. But keep thinking forward. ACRS wrote a letter saying that transients in passive plans have to be analyzed all the way to when you reach a stable condition, whether it's 2 hours, 24 hours, or 150 hours.

If you are continuously degrading and getting into a less safe condition, you should not artificially stop at 72. That's our position and maybe go against regulation. But certainly the logic tells us that's what one should do.

MEMBER REMPE: On the other hand, it's for mitigating actions could occur which should be within 72 hours. And the fact that it's still not a stable state, I'm not sure your logic holds.

CHAIR MARCH-LEUBA: No, what I'm saying you have analyze it.

MEMBER REMPE: But if they want longer than 72 hours, yeah, it ought to be longer. But if you can say, I can get my --

(Simultaneous speaking.)

CHAIR MARCH-LEUBA: I'm not against getting credit for mitigation actions past 72 hours.

MEMBER REMPE: Okay.

CHAIR MARCH-LEUBA: You don't have any Class 1E power. But you are going to get power in 72 hours.

MEMBER REMPE: Yeah, and that was the reason. So I was having trouble following your logic. But if you want longer for mitigating strategies to be implemented, then I get what you're saying. But anyhow, I just had to --

MR. SHEA: Actually, it's kind of interesting because I researched this extensively. I was -- anybody on the phone?

MR. NOLAN: Yeah, this is Ryan Nolan from the staff. I was going to try to add some context. But go ahead, Jim. You go first.

MR. SHEA: Okay. I was looking at this myself because I was curious on the 72 hours. And it's not really in any regulation. And when I was thinking about it, there's a lot of other things come into play when you're talking about events, including RTNSS. And so when you get the whole design and they talk about all the mitigation that's going to come to play, including then mitigating strategies.

For a design basis case, it's a specific thing. It's artificial in a way. And that's what they have to do to meet these LOCA regulations in the DBA. So that's what kind of comes into play in my mind.

CHAIR MARCH-LEUBA: Let me say something before you go. We have a tendency of concentrating a lot because that's what we do on Chapter 15 design basis events. My comment was more related to Chapter 19, low probability events.

And we kind of tend not to think about those for some reason simply because we always do Chapter 15. That's what our job is. That's how we've been doing all this life. And we need to think beyond Chapter 15. I think Shanlai wanted to say something or Joe.

MR. STAUDENMEIER: Yeah, I was going to say the history --

CHAIR MARCH-LEUBA: Your name?

MR. STAUDENMEIER: Joe Staudenmeier. The history of that 72 hours, it was something that came up with some utility requirements that EPRI put together something saying for the passive plants, like, the first two were ESBWR and they peak at 600. And the Commission endorsed those requirements at the time saying, yes, these are good things to meet. But they're not really regulation. But it was guidelines that the passive plants would have to survive for 72 hours.

CHAIR MARCH-LEUBA: As I said before, it's not relevant to what we're discussing today. But it will become eventually.

MR. NOLAN: Yeah, so this is Ryan Nolan with the staff. I'll just add that when we do receive an application, the Commission policy not only is it 72 hours with no operator actions. But it's also cope onsite for 7 days for all design basis events.

And so the staff will look at a minute a 7-day period of time. You just get additional flexibility under the regulatory treatment of non-safety system policies for what systems you can use and what quality those systems need to be post-72-hours and post-7 days. And so, yeah, we agree with you. You have to remain in a safe stable condition.

It's just after 72 hours, there's additional flexibility for what you're allowed to use in your analysis. And whether that analysis ultimately ends up in Chapter 15 or 19, that's a virtual cost when we get the application. But it's certainly something we'll be looking at.

MR. LU: I think Ryan has already covered that one specifically. So it will be handled by

Chapter 19 specifically, the partition between PRA branch and the reactor system branch. But the 72 hour as Joe mentioned, it was initially proposed by EPRI and endorsed by the Commission with the SRM for 72 hours. But I think that Ryan mentioned that we are in alignment with ACRS for the long term mitigation beyond the 72 hours and the readiness program there. But it will be reviewed by Chapter 19.

CHAIR MARCH-LEUBA: Okay. Thank you to staff for this discussion.

MEMBER HALNON: Yeah, and this is Greg. For the record, that covers my question on the small break LOCA. I will hold it till Chapter 15 and 19.

CHAIR MARCH-LEUBA: I don't see any raised hands. So let's continue.

MEMBER DIMITRIJEVIC: I just also want to mention that in Chapter 19, simply the mission time for the large release is 72 hours. That changed simply just for the new plans from 44. And I couldn't find the base for the change. I mean, I was trying to see how that change occurred from 24 hours to 72. But what Joy said, this is after 72 hours. You can apply different mitigation measures.

CHAIR MARCH-LEUBA: Thank you, Vesna.

MR. HAIDER: This is Syed Haider back.

This slide provides the two containment specific limitations and conditions number 3 and 4 that the staff imposed as a result of the staff review. According to the limitation and condition number 3 as mentioned earlier in the presentation, limitation and condition number 3 merged out of the potential degradation of the isolation condenser heat transfer performance should there be a reverse flow with one condenser from the containment to the RPV.

Such a flow could take place if PCCS is not properly sized to deal with the most limiting flow reversal condition. Therefore, the licensing stage, the applicant will need to demonstrate that either no reverse flow would occur or any reverse flow that occurs under the most bounding flow reversal conditions is not safety-significant with respect to the methodologies acceptance criteria. Regarding limitation and condition number 4 requires that if any alternate PCCS design configuration and placement is used for the BWRX-300 design and the licensing basis stayed other than the one presented in this LTR, the applicability of this method. And the PCCS modeling approach would have to be review and approved by the NRC.

CHAIR MARCH-LEUBA: I made the same

comment earlier this morning, but you guys were not here. On safety evaluation reports, language makes a difference. And there is a difference between the word review and the word audited.

So if your intention is to audit the new PCCS, that means you have an approval to keep using TRACG-GOTHIC for this analysis unless we find something wrong. If you say, it must be reviewed and approved, it kind of implies that you must issue a new SER or a revision of the SER for the new PCCS. What is your intent?

MR. HAIDER: The intent is a review.

CHAIR MARCH-LEUBA: Review and approve means you must issue an SER. And you can review it in an audit in which case it's by exception. You have approval to do it unless we tell you otherwise.

But if you say must be reviewed and approved, you don't have approval to do it until we send you documentation that you can. So language makes a difference. And you need to find out what is it that you are going to do.

MR. HAIDER: Okay. I think we can probably discuss some of the detail that might be relevant.

CHAIR MARCH-LEUBA: I'm just putting it in your brain so you understand the difference between you are approved to do it unless we tell you otherwise versus you cannot do it until I give you documentation that you can. It's a big difference.

MR. HAIDER: Okay.

MEMBER REMPE: Am I hearing that you have already or you're already aware of this difference and you want to review it? Or you want to think about it more?

MR. HAIDER: The language that we have crafted in the limitation and condition number 4 is review and approve.

MEMBER REMPE: And that's what you want.

MR. HAIDER: I'm not willing to discuss the design details. They have commented on what kind of changes they might be implementing in the PCCS design.

CHAIR MARCH-LEUBA: My concern, I'm always thinking what can possibly go wrong. We might be making our life difficult by using the wrong wording because a safety evaluation report has -- I mean, it's not low. But the language has weight. You must define.

MR. SHEA: This is Jim Shea again. I think the key to all these LTRs is that there will be a final review -- a staff review for a licensing action. These aren't licensing actions. They're pre-application LTRs that we're using as a means to approve, like in this case, a methodology.

And the methodology would be used no matter what the PCCS final design is and that's what we're saying. But however, like Syed said, he would open that up to an audit to see that the methodology is still applying to that newer design and overall that the SE for the final licensing action would be the closure.

MEMBER REMPE: So to make sure I understand, you're saying if they put it in a construction permit or a Part 52 type of application that the final review and approval will be the staff SE on that.

MR. SHEA: Correct.

MEMBER REMPE: Okay.

MR. SHEA: Yeah.

MR. HAIDER: Okay. Thank you. Now finally we will present the staff conclusions for our LTR review. In summary, the proposed BWRX-300 analytical approach, and the TRACG and GOTHIC modeling of mass-energy release and containment response are acceptable with the appropriate conservative biases demonstrated in the LTR. With the four Limitations and Conditions as specified in the staff SER, the analysis method presented in the LTR Revision 2 is acceptable for BWRX-300 containment pressure and temperature analysis of the design basis accidents.

The NRC staff will be required to evaluate the regulatory compliance of the final BWRX-300 containment design using the containment evaluation methodology during the future licensing activities, in accordance with 10 CFR Part 50 or 10 CFR Part 52, as applicable to the application. And this concludes our presentation. Thank you very much for attending. Now the staff would like to address any additional questions the Committee still might have.

CHAIR MARCH-LEUBA: And in spite of all my best efforts, we are still on time. So we do have time for a couple of questions. If the members or Vesna want to ask anything from the staff?

(Simultaneous speaking.)

CHAIR MARCH-LEUBA: Anything else? Vesna, you said I don't?

MEMBER DIMITRIJEVIC: Yeah, I said I'm good.

CHAIR MARCH-LEUBA: Thank you. I don't see any questions from the staff. I am going to open the floor for public comments. So at this point, I would like to open the session to comments from the public.

You are attending the MSTeams. Just unmute yourself. If you're calling on the bridgeline, unmute via *6. Just a reminder that this is an opportunity to add comments to the public record for Committee consideration.

If you have any questions or particular requests, please address them directly to our DFO, Mike Snodderly. His email is michael.snodderly@nrc.gov. Any public comments, please say your name and give us your comments.

(No audible response.)

CHAIR MARCH-LEUBA: I don't hear anybody. So at this point, we're going to recess for 15 minutes until 3:00 o'clock Eastern and in which case we will start the closed session. So anybody that belongs in the closed session has a link.

Please call as soon as possible so the staff and GEH can confirm that you can join. And this always takes some time. So I will recommend that you join the session now, put yourself on mute, and take the break after that. So we will reconvene on the closed session at 3:00 o'clock Eastern Time in 15 minutes. This meeting is recessed and we will not be going back to this line.

(Whereupon, the above-entitled matter went off the record at 2:45 p.m.)

HITACHI

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M220009
March 9, 2022

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Canadian Nuclear Safety Commission
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Subject: ACRS Subcommittee Open Session Presentation Slides for NEDC-33922P, BWRX-300 Containment Evaluation Method Licensing Topical Report

Enclosed are the final presentation slides that GE Hitachi Nuclear Energy (GEH) will present during the Open Session at the upcoming Advisory Committee on Reactor Safeguards (ACRS) subcommittee meeting on March 18, 2022. These slides support the ACRS review of NEDC-33922P Revision 2, BWRX-300 Containment Evaluation Method, and the corresponding Advanced Safety Evaluation Report (SER) with No Open Items.

Enclosure 1 contains non-proprietary information and may be made available to the public.

If you have any questions, please contact me at 910-200-3295.

Sincerely,

George E. Wadkins

George E. Wadkins
Vice President, New Power Plants and Products Licensing
GE-Hitachi Nuclear Energy Americas, LLC

Enclosure:

1. ACRS Subcommittee Open Session Presentation Slides for NEDC-33922P, BWRX-300 Containment Evaluation Method Licensing Topical Report – Non-Proprietary Information

Document Components:
001 M220009 Cover Letter.pdf
002 M220009 Enclosure 1 Non-Proprietary.pdf

cc: James Shea, US NRC
Chantal Morin, CNSC
PLM Specification 006N9430 Revision 0

ENCLOSURE 1

M220009

ACRS Subcommittee Open Session Presentation Slides for
NEDC-33922P,
BWRX-300 Containment Evaluation Method Licensing Topical Report

Non-Proprietary Information

ACRS Subcommittee Presentation

GE-Hitachi (GEH)
Licensing Topical Report (LTR) NEDC-33922P
BWRX-300 Containment Evaluation Method

(Open Session)

March 18, 2022

Agenda

- Licensing Topical Report Purpose and Scope
- Acceptance Criteria for Containment Evaluation Method
- RPV and Containment Features Pertinent to the Method
- LOCA Scenarios and Limiting Pipe Breaks
- Overview of the Evaluation Model
- TRACG Method for Mass and Energy Release
- Containment Analysis Method Using GOTHIC

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Licensing Topical Report Purpose and Scope

LTR Purpose and Scope (LTR Section 1.1)

GEH is seeking NRC approval for application of an analysis method to be used for evaluating the BWRX-300 dry containment thermal hydraulic performance.

The LTR scope includes:

- Method description
- Method qualification
- Sensitivity studies
- Application of the method to the BWRX-300 for the events identified above
- Demonstration cases

LTR Purpose and Scope (LTR Section 1.1)

The analysis method to be used for the BWRX-300 containment thermal hydraulics performance demonstrates that the containment design complies with the following acceptance criteria listed in Section 4.0 of Licensing Topical Report (LTR) NEDC-33911P, BWRX-300 Containment Performance:

- The containment pressure boundary and penetrations are designed for the design pressure and temperature to be established for design basis accidents (DBAs) during future licensing activities in accordance with 10 CFR 50, Appendix A, General Design Criteria (GDC) 2, 4, 16, 38, 41, 50, and 51.
- In accordance with 10 CFR 50, Appendix A, GDC 4, 16, 38, 41, 50, and 51, containment design pressure will be evaluated during future licensing activities to bound the peak accident containment pressure resulting from the most limiting large break loss-of-coolant accident (LOCA) with margin, with no less than 10% margin during the preliminary safety analysis report (PSAR) phase in order to conform to Standard Review Plan (SRP) 6.2.1.1.A Acceptance Criteria.
- In accordance with 10 CFR 50, Appendix A, GDC 16, 38, and 50, the BWRX-300 containment design features establish an essentially leak-tight barrier, and will be demonstrated during future licensing activities to reduce containment pressure and temperature rapidly, and maintains them at acceptably low levels following a LOCA; and the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from a LOCA.

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LTR Purpose and Scope (LTR Section 1.1)

- The methodology described in the LTR complies with all applicable regulatory requirements as written.

- GEH is not requesting NRC approval for exemptions from any regulatory requirements.

Technical Evaluation

Containment Design Acceptance Criteria (LTR Section 1.3)

- Accident pressure and temperature are less than design pressure and temperature with appropriate margin
- Containment pressure is reduced to less than 50% of the peak accident pressure for the most limiting LOCA within 24 hours
- Containment pressure responses after 24 hours for LOCAs that do not produce the peak accident pressure are maintained below 50% of the peak pressure for the most limiting LOCA
- Containment atmosphere remains sufficiently mixed such that deflagration or detonation does not occur inside containment

The containment evaluation method presented in the topical report is an acceptable method to demonstrate that the above performance design criteria are met.

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Relevant RPV and Containment Design Features (LTR Section 2.0)

The RPV, including the isolation condenser, piping and RPV isolation valves, are described in NEDC-33910P, “RPV Isolation and Overpressure Protection”

Conceptual containment design, penetrations, isolation valves, and Passive Containment Cooling System (PCCS) are described in NEDC-33911P, “Containment Performance”

The following containment design features are relevant to the purposes of the LTR:

- Containment is a dry enclosure, near atmospheric pressure during normal operation
- Containment design pressure and temperature are within the experience base of conventional BWRs
- Containment is inerted with nitrogen during normal operation
- There are no subcompartments containing large bore high energy lines
- The subcompartments have sufficiently large openings such that the boundaries of the subcompartments do not experience large pressure differentials resulting from pipe breaks outside the subcompartments

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LOCA Scenarios and Limiting Pipe Breaks (LTR Section 3.0)

- All large bore piping attached to the RPV has 2 isolation valves. The RPV isolation valves, except for those on the IC piping, close on either of the following conditions:
  - High drywell pressure
  - Low-low RPV water level
- An assumed single failure rendering an ICS train inoperative is the most limiting single failure for small breaks because the small breaks rely on ICS to depressurize the RPV.
- The limiting large breaks are:
  - Main steam pipe
  - Feedwater pipe
- All design basis large breaks are rapidly isolated at the RPV nozzle.
- The limiting small breaks are unisolated instrument line breaks, either in the steam or liquid space.

Overview of the Evaluation Model

Overview of the Evaluation Model (LTR Section 4.0)

TRACG is used to evaluate the mass and energy release from the BWRX-300 RPV

- Utilizes applicable parts of TRACG Application for ESBWR approved LTR, which is incorporated in the approved ESBWR Design Certification
- Section 5.0 of the LTR details application of the ESBWR TRACG method to the BWRX-300.

GOTHIC is used to evaluate the BWRX-300 containment response

- New containment model developed for BWRX-300
- Based on GOTHIC computer code – meets 10 CFR 50 Appendix B QA requirements
- The GOTHIC code has been benchmarked to separate effect and integral tests. Benchmarking to the test data of a similar size containment is included in the LTR.

Evaluation method for the BWRX-300 containment response to design basis events developed per the applicable elements of RG 1.203.

- Uses base and conservative cases
- Individual key inputs, assumptions and modeling parameters conservatively biased simultaneously in the conservative cases (same approach taken for the ESBWR containment method in NEDC-33903P-A Revision 1, “TRACG Application for ESBWR”)
- Evaluation method described in LTR Section 6.0.

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TRACG Method for Mass and Energy Release (LTR Section 5.0)

- TRACG Code and Qualification (LTR Section 5.1)
  - TRACG is the GE Hitachi Nuclear (GEH) proprietary version of the Transient Reactor Analysis Code (TRAC).
  - TRACG uses realistic 1D and 3D models and numerical methods to simulate phenomena that are experienced in the operation of boiling water reactors (BWRs)
  - TRACG ECCS-LOCA analysis method for BWR/2–6 and for ESBWR have been approved previously.
  - The uncertainties in the TRACG ECCS-LOCA analysis methods are quantified for the BWR/2-6 and for ESBWR
    – Only the uncertainties relating to RPV inventory and break flow are accounted for in the BWRX-300 application

TRACG Method for Mass and Energy Release (LTR Section 5.0)

- Applies the ESBWR TRACG-LOCA method to the BWRX-300 mass and energy release calculations (LTR Section 5.2)
  - TRACG RPV Nodalization for BWRX-300 (LTR Section 5.2.1) – For RPV and ICS, more detailed than ESBWR
  - Evaluation of Large and Small Steam and Feedwater Pipe Breaks (LTR Section 5.2.2)
    - The main steam piping of both loops up to the turbine stop valves are included in the model, and a large steam pipe break flow includes the flow from both ends of the break conservatively assuming instantaneous separation of the pipes.
  - Channel Grouping, Decay Heat, and Power Shape (LTR Section 5.2.3)
  - Modeling of the Isolation Condenser and Radiolytic Gases (LTR Section 5.2.4)
  - Modeling Biases (PIRTs) (LTR Section 5.2.5)
    - Conservative biases (LTR Table 5-1) are used to bound uncertainties in key TRACG models
    - For conservative cases, the initial operating conditions are also conservatively biased (LTR Table 5-4)
  - Initial Conditions for Conservative Cases: Trips and Isolation Signals bound the time required to reach the analytical setpoint, signal development time, and the time required for valve stem travel until the area restriction starts to occur. (LTR Section 5.2.6)

TRACG Method for Mass and Energy Release (LTR Section 5.0)

GEH performed both base and conservative demonstration cases:

- Demonstration Cases (Base and Conservative) for Main Steam and Feedwater Large Breaks (LTR Sections 5.3, 5.3.1, 5.3.2)
- Demonstration Cases (Base and Conservative) for Small Steam and Liquid Pipe Break Cases (LTR Section 5.4)

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Containment Analysis Method Using GOTHIC (LTR Section 6.0)

The GOTHIC application methodology includes base cases and conservative cases.

- Base Cases:
  - Nominal inputs, assumptions, and correlations

- Conservative Cases:
  - Individual key inputs, assumptions and modeling parameters are conservatively biased simultaneously
  - This method compounds conservatisms and provides reasonable assurance that the overall method results bound the uncertainties.

- This is the same approach that was taken for the ESBWR containment method.

Details of Containment Analysis Method Using GOTHIC (LTR Section 6.0)

- Identification of the relevant inputs and phenomena relevant to the BWRX-300 containment response and the selection of the models and correlations used to develop the base GOTHIC containment model (LTR Sections 6.1 – 6.4)
- GOTHIC input model for the BWRX-300 containment (LTR Section 6.5)
- Base cases and the results obtained from those base cases (LTR Section 6.6)
- How nodalization impacts the calculated results (LTR Section 6.7)
- Which model uncertainties and biases of the GOTHIC methodology are most important for application of the GOTHIC model for analyses of the BWRX-300 containment (LTR Sections 6.5 through 6.7)
- Key model uncertainties and biases used in developing the conservative GOTHIC containment model (LTR Section 6.8)
- Benchmark predictions of test data (LTR Section 6.9)
- Demonstration analyses showing the BWRX-300 containment response for various break sizes and locations using the conservative GOTHIC containment model (LTR Section 6.10)
  - Large steam line break (LTR Section 6.10.1)
  - Small steam or liquid breaks (LTR Section 6.10.2)
- Summary of assumptions and inputs used in the GOTHIC conservative cases (LTR Section 6.11)

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Conclusion

In summary…

- The methodology described in the LTR complies with all applicable regulatory requirements as written.
- GEH is not requesting NRC approval for exemptions from any regulatory requirements.
- TRACG utilizes the applicable parts of the TRACG Application for ESBWR approved LTR, which is incorporated in the approved ESBWR Design Certification
- Utilizes GOTHIC, a standard code used for evaluating thermal-hydraulic containment response in the nuclear industry
- Individual key inputs, assumptions and modeling parameters conservatively biased simultaneously in the conservative cases (same approach taken for the ESBWR containment method)

Questions or Comments

Regulatory Review of GEH Topical Report "BWRX-300 Containment Evaluation Method"

NEDC-33922P, Revision 2

NRC Staff Presentation-Open Session

BWRX-300 Small Modular Reactor
ACRS Subcommittee Meeting

March 18, 2021

GEH BWRX-300 Pre-Application Status

- September 26, 2019 – BWRX-300 Pre-Application Kick of Meeting
- 3 LTRs have been approved

  Reactivity Control LTR (NEDC-33912P) (04/01/21)
  Reactor Pressure Vessel Isolation and Overpressure Protection LTR (NEDC-33910P) (08/16/21)
  Containment Performance LTR (NEDC-33911P) (01/07/22)

- 2 LTRs currently under review

  Containment Evaluation Method LTR (NEDC-33922P) (AFSE 2/18/22)
  Advanced Civil Construction and Design Approach LTR (NEDC-33914P) (AFSE 2/18/22)

- 3 LTRs expected in 2022

  Safety Strategy (NEDC-33934P) 2Q22
  Severe Accident Management and Source Term Methodology (NEDC-33913P) 3Q22
  Instrumentation and Controls Architecture (NEDC-33925P) 3Q22

NRC Staff Review Team

- NRR Nuclear Systems Performance Branch (SNSB)
  - Syed Haider
- NRR New Reactor Licensing Branch (NRLB)
  - James Shea
- NRR Nuclear Methods, Systems & New Reactors Branch (SNRB)
  - Carl Thurston, Shanlai Lu
- NRR Containment and Plant Systems Branch (SCPB)
  - Chang Li
- RES Code and Reactor Analysis Branch (CRAB)
  - Peter Lien, Joe Staudenmeier, Andrew Ireland
- RES Fuel & Source Term Code Development Branch (FSTCB)
  - Shawn Campbell

Presentation Outline

- Overview of BWRX-300 Containment Evaluation Method (CEM) LTR NEDC-33922P, Revision 2
- BWRX-300 containment design background
- Regulatory requirements & BWRX-300 acceptance criteria for containment response
- BWRX-300 containment evaluation method demonstration analyses
- TRACG mass and energy release calculation methodology review
- GOTHIC containment response calculation methodology review
- Resulting four limitations and conditions
- Conclusions

NRC Staff Review of the LTR

- The purpose of GEH LTR NEDC-33922P, Revision 2, is to obtain NRC staff approval of the BWRX-300 containment peak pressure and temperature analysis methodology.

- The NRC regulations and acceptance criteria dealing with the BWRX-300 containment thermal hydraulics performance are referenced in GEH LTR NEDC-33911P, BWRX-300 Containment Performance.

- The approved methodology will be used to design the BWRX-300 containment and support a license application for a CP and OL under 10 CFR 50 or a DCA and COL under 10 CFR 52.

BWRX-300 Containment Design Background

- BWRX-300 has a nitrogen-inerted, dry containment
- No suppression pool inside the containment
- RPV isolation valve closure limits M&E release in LBLOCA
- RPV remains unisolated for SBLOCA with break flow
- Passive Containment Cooling System (PCCS)
  - Long-term containment SBLOCA pressure mitigation
  - Demo with specific LTR described units
- Reactor cavity pool for containment heat removal
- Containment dome interfacing with the reactor cavity pool

Regulatory Requirements & BWRX-300 Acceptance Criteria for Containment Response

- Key Regulatory Requirements
  - Short term peak containment pressure/temperature
  - Long term pressure/temperature
- BWRX-300 LTR Acceptance Criteria
  - Accident pressure and temperature are less than design pressure and temperature with appropriate margin
  - Containment pressure is reduced to less than 50% of the peak accident pressure for the most limiting LOCA within 24 hours
  - Containment pressure responses after 24 hours for LOCAs that do not produce the peak accident pressure are maintained below 50% of the peak pressure for the most limiting LOCA
  - Containment atmosphere remains sufficiently mixed such that deflagration or detonation does not occur inside containment

BWRX-300 Containment Evaluation Method Demonstration Analyses

- Containment analysis method for BWRX-300 thermal-hydraulic performance is used to demonstrate that the containment design satisfies the acceptance criteria for:
  - Large-Break Loss-of-Coolant Accident (LBLOCA)
  - Small-Break Loss-of-Coolant Accident (SBLOCA)

- Analyzed containment DBEs include liquid and steam breaks

- TRACG code is to calculate the mass and energy release and GOTHIC code to calculate the containment response

- Acceptance criteria were satisfied for the LTR demonstration cases

TRACG Code - Overview for BWRX-300

- Overview of TRACG code
  - ➢ Latest TRACG versions used in analysis, no significant changes since ESBWR
  - ➢ RPV model and internal components scaled from ESBWR
  - ➢ De-coupled method assumes Containment remains at atmospheric pressure

- Past TRACG approval and relevance to BWRX-300
  - ➢ ESBWR qualification extended to BWRX-300, such that ESBWR PIRT and model biases applied for RPV and internals
  - ➢ BWR/2–6 methods evoked since some events result in core uncover
  - ➢ IC's safety function changed and modeled in considerably more detail
  - ➢ Modeling deemed adequate for M&E release calculations (w/ L&Cs applied)

TRACG Code – Mass and Energy Release Calculation Methodology

- BWRX-300 unique design features in comparison with ESBWR
  - LBLOCA isolation (Previous Approved LTR)
  - No suppression pool
  - ICs are the primary decay heat removal path
- RPV isolation valves limits break flow and M&E release for large piping but small breaks are un-isolated and continue blowdown for 72 hours
- One ICS train inoperative (due to limiting single failure)
- Conservative inputs for initial power level, power history, scram time, choke flow model, atmospheric pressure break boundary condition and bounding operating conditions

TRACG Code – Mass and Energy Release Calculation Methodology

Significant Issues and Resolution

RAI – Radiolytic gas accumulation and removal in the ICs

L&C 1: total volumetric fraction of radiolytic gases in the IC lower drum limited to a sufficiently low level such that condensation heat transfer in the ICs is not adversely affected and the hydrogen deflagration margin is maintained

RAI – ICs return line steam trap

L&C 2: IC return line layout must include a loop seal, or water trap, that prevents reverse flow from RPV back into the IC return line

GOTHIC Code Overview for BWRX-300

- Overview of GOTHIC code
  - An established industry code widely used in the containment response analysis
  - 10 CFR Part 50, Appendix B compliant code
  - Latest GOTHIC version 8.3 used in the BWRX-300 analysis
- Past GOTHIC approval and relevance to BWRX-300
  - GOTHIC previously approved for containment response analysis
  - BWRX-300 containment PIRT consistent with GOTHIC functionalities
  - BWRX-300 relevant GOTHIC benchmarking against CVTR test data reviewed
  - GOTHIC is qualified for the thermal and species stratification and 3D effects

GOTHIC Code - Containment Response Calculation Methodology

- Based on Reg Guide 1.203, "Transient and Accident Analysis Methods"
- Decoupled M&E release from the TRACG RPV model with no backpressure as a containment BC for the stand-alone GOTHIC containment model
- 4-component GOTHIC model
  - Containment (nodalized)
  - Dome (nodalized)
  - PCCS (nodalized)
  - Reactor Cavity Pool (lumped)
- Conservative Diffusion Layer Model (DLM) used for condensation
- Thermal stratification inside the containment

Staff Review of the BWRX-300 GOTHIC Containment Response Methodology

- Physical phenomena (GOTHIC PIRT)
- GOTHIC input model (Nominal inputs, assumptions, and correlations)
- Key modeling uncertainties and conservative biases -- Overall GOTHIC model conservatism
- Nodalization sensitivity studies for the containment and PCCS
- Benchmark predictions of test data
- BWRX-300 containment response analyses for large/small breaks
- PCCS capacity to mitigate the containment pressure in the long-term.
- Containment mixing for combustible gases
- Staff confirmatory analyses

Significant Containment-specific Issues & Resolutions

- Break location & and break flow orientation sensitivities
  - Limiting PCP LBLOCA location and orientation modified
  - A liquid, and not steam, SBLOCA is limiting
- Sensitivity to containment nodalization
  - Potential for reverse flow for SBLOCA and non-condensable gas return to RPV
  - L&C #3 – No break flow reversal
- Containment heat transfer modeling
  - Identification of the PCCS condensation modeling error.
- PCCS modeling and nodalization sensitivity study
  - L&C #4 – Applicability to the final PCCS design for licensing basis

RPV-specific Limitations and Conditions

- L&C #1

The use of this CEM is limited to a BWRX-300 design that limits the total volumetric fraction of radiolytic gases in the IC lower drum to a sufficiently low level throughout a 72-hour period following the event such that condensation heat transfer in the ICs is not adversely affected and the hydrogen deflagration margin is maintained

- L&C #2

The use of this CEM is limited to a BWRX-300 design that a proper isolation condenser return line layout is chosen, such as a loop seal or a water trap, to prevents reverse flow from RPV into the IC return line throughout a 72-hour period following the event or where an applicant or licensee referencing this report demonstrates that the TRACG code is capable of conservatively modeling the overall ICs heat removal capacity when reverse flow occurs in the IC discharge lines.

Containment-specific Limitations and Conditions

- L&C #3.

The use of this CEM is limited to a BWRX-300 design in which the PCCS is sized sufficiently large such that a reverse flow from containment back to RPV does not occur during the first 72-hours into the event. The applicant or licensee referencing this report needs to demonstrate that no reverse flow could occur, or any reverse flow that occurs under the most bounding flow reversal conditions resulting in the degradation of IC heat transfer is not safety-significant with respect to the acceptance criteria for the BWRX-300 CEM.

- L&C #4.

The use of this CEM was demonstrated for a BWRX-300 design with the PCCS described in this LTR. For any alternate PCCS design configuration and placement, the applicability of this method and the PCCS modeling approach must be reviewed and found to be acceptable by the NRC for BWRX-300 licensing-basis analyses.

Conclusions

- The proposed BWRX-300 analytical approach, and TRACG/GOTHIC modeling described in the LTR for M&E release and containment response are acceptable, with the appropriate conservative biases and modeling inputs to address the model uncertainties.

- With the four Limitations and Conditions specified in the staff SER Section 7.0, the NRC staff concludes that the evaluation methodology presented in GEH LTR NEDC-33922P, Revision 2, is acceptable for BWRX 300 containment peak containment pressure and temperature analysis of the containment design basis accidents.

- The NRC staff will evaluate the regulatory compliance of the final BWRX-300 containment design using the CEM during the future licensing activities, in accordance with 10 CFR Part 50 or 10 CFR Part 52, as applicable.

Meeting Summary

Meeting Title	ACRS Subcommittee Meeting - Containment Evaluatio
Meeting Start Time	3/18/2022, 11:56:56 AM
Meeting End Time	3/18/2022, 3:44:56 PM
Meeting Id	cdbd34f7-2b0d-47aa-8257-8272a09baa49

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