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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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667TH MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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WEDNESDAY

OCTOBER 2, 2019

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ROCKVILLE, MARYLAND

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The Advisory Committee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2D10, 11545 Rockville Pike, at 1:00 p.m., Peter Riccardella, Chairman, presiding.

COMMITTEE MEMBERS:

PETER RICCARDELLA, Chairman

DENNIS BLEY, Member

CHARLES H. BROWN, JR., Member

WALTER L. KIRCHNER, Member

JOSE MARCH-LEUBA, Member

DAVID PETTI, Member

JOY L. REMPE, Member

DESIGNATED FEDERAL OFFICIAL:

QUYNH NGUYEN

ZENA ABDULLAHI

ALSO PRESENT:

JAMES (ALAN) BEARD, GEH

YOUSEF FARAWILA, Framatome

JANE MARSHALL, NRR

JASON PAIGE, NRR

WALTER (SKIP) SCHUMITSCH, GEH

JAMES SHEA, NRO

ASHLEY SMITH, NRR*

DANIEL TINKLER, Framatome

*Present via telephone

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P R O C E E D I N G S

(1:03 p.m.)

CHAIRMAN RICCARDELLA: This meeting will now come to order. This is the first day of the 667th meeting of the Advisory Committee on Reactor Safeguards. I'm Pete Riccardella, Chairman of the ACRS. The ACRS was established by the Atomic Energy Act and is governed by the Federal Advisory Committee Act, FACA.

The ACRS section of the U.S. NRC public website provides information about the history of the ACRS and provides FACA-related documents, such as charter, bylaws, Federal Register notices for meetings, letter reports, and transcripts of all full and subcommittee meetings, including all slides presented at the meetings.

The Committee provides its advice on safety matters to the Commission through its publicly available letter reports. The Federal Register notice announcing this meeting was published on September 18, 2019, and provided an agenda and instructions for interested parties to provide written documents or request opportunities to address the committee, as required by FACA. In accordance with FACA, there is a designated federal official for today's meeting.

The DFO for this meeting is Mr. Quyhn Nguyen.

During today's meeting, the Committee will consider the following: advanced boiling water reactor design certification renewal and Framatone's topical report, RAMONA5 for anticipated transient without SCRAM, and, three, preparation of ACRS reports.

There is a phone bridge line. To preclude interruption of the meeting, the phone will be placed in a listen-only mode during the presentations and committee discussions. We have received no written comments or requests to make oral statements from members of the public regarding today's session.

There will be an opportunity for public comment, as we have set aside ten minutes in the agenda for comments from members of the public attending or listening to our meeting. Written comments may be forwarded to Mr. Quyhn Nguyen, the designated federal official.

A transcript of open portions of the meeting is being kept, and it is requested that all speakers use one of the microphones, identify themselves, and speak with sufficient clarity and volume so that they can readily be heard. Also, please silence any phones or other devices to avoid interruption of the meeting. We're somewhat

short-handed this week, as we have two members overseas, one sick, and one not attending, but we'll have adequate coverage of the topics based on the members who are in attendance.

Also, I'd like to acknowledge the passing of a former ACRS member and chairman, Dr. Mario Bonaca. He served with the Committee until 2014. The first topic at the meeting is ABWR, advanced boiling water reactor design certification renewal.

This is the -- we've done many design certifications, but this is the first one to come up for renewal. With that, I will turn the meeting over to Jason Page, who is acting branch chief of Licensing Branch 3 in NRO. Jason.

MR. PAGE: Thank you. I'm just going to introduce myself again. My name is Jason Page, acting branch chief in the office of new reactors. I'll turn it over to Jim Shea. He's the PM of this activity.

MR. SHEA: Thanks, Jason. Again, I'm Jim Shea. Good afternoon, everyone. I appreciate the opportunity for us to present our staff review of the ABWR D.C. renewal. GEH will go first, and then the staff will follow. It will be an abbreviation of what we did before for the subcommittee. Thanks. I would like to turn it back over the Chairman.

MR. BEARD: Good afternoon. My name is Alan Beard. I'm with GE-Hitachi Nuclear Energy. With me is Walter "Skip" Schumitsch. He's the program manager for this effort to renew our certification for the ABWR.

We're just going to very quickly go over what we've done as part of the renewal process. Just a quick overview of what the ABWR is, for those of you who are not real familiar with it, a brief discussion of the timeline that we've been through through the renewal effort, the scope of what we did during that renewal effort, and then just a real quick talk about some of the major design changes that we made as part of that process.

ABWR was first built and operated in Japan, at the Kashiwazaki-Kariwa site. Units 6 and 7 are both -- were in operation, are currently suspended in operation because of the events of the tsunami, but there are plans to bring those back online. Japan has three additional ABWRs that were operating prior to the tsunami. As far as we know, there are plans to bring those back online. Additional two are under construction in Japan, and then two under construction in Taiwan, although the Taiwan construction effort is currently suspended. ABWR is licensed in Japan and

Taiwan, certified in the U.S., and underwent the approval process, GDA approval process in the United Kingdom, so it's had a lot of review.

CHAIRMAN RICCARDELLA: Did the plants at KK, did they operate before the tsunami?

MR. BEARD: Oh, yes, they had about 25 years of operation between the two of them.

CHAIRMAN RICCARDELLA: Twenty-five years.

MR. BEARD: Yes. I would also note that the ABWR, at least in our opinion, is the first of the Generation 3 reactors that has been in operation.

MEMBER REMPE: Out of curiosity, because I missed your subcommittee meeting in -- educate me. The ones in Japan and other ones that are up and were running, do they operate in the MELLLA+ region or MELLLA region? Do they just use control rods for power changes, or do they use flow, also?

CHAIRMAN RICCARDELLA: Do you know?

MR. SCHUMITSCH: I do not -- sorry, this is Skip Schumitsch. I'm sorry; I do not know the answer to that, either.

MEMBER REMPE: It's probably not relevant. I just was curious. Do you know, Jose, from your work on it?

MEMBER MARCH-LEUBA: From the original

reviews of ABWR, they do use flow for power control, yes.

MEMBER REMPE: So they never have gone to any sort of expanded operating domain, then?

MEMBER MARCH-LEUBA: I suspect it's the same with ABWRs and BWRs -- I don't know -- that they have such high power density that there's no need to go to MELLLA+. They're already a DPU when they were licensed.

MEMBER REMPE: Okay, thank you.

MR. BEARD: Would also like to note -- it's not on this slide, but the ABWR --

MR. SCHUMITSCH: I'm sorry; we got a text from somebody that's listening. The answer is not MELLLA+. Thank you, David.

MR. BEARD: -- to note that the ABWR was the first plant that underwent the Part 52 process and was issued the No. 1 certification, Appendix A of Part 52. That was back in May of 1997. I will note, there's a picture over on the wall there. Our initial meetings with the ACRS during that ABWR certification were actually held down in the green building, off of Norfolk Avenue, in Bethesda, when the ACRS was still meeting down there. There are names on the table there of people that were actually part of the review

initially. Then one other note, just kind of a historical interest.

We actually submitted the ABWR under Part 50, originally, seeking a standard safety analysis report, then Part 52 became promulgated, became a regulation, and we chose to take advantage of that, so we switched our application over to a Part 52 application.

Just like to note that the ABWR was a collaborative effort. It was developed in between the efforts of GE, Tokyo Electric Power Company, Hitachi, and Toshiba. As I noted before, the first plants that were built and operated, Kashiwazaki-Kariwa Units 6 and 7.

We'll note that both of those plants were built on time and ahead of schedule and under budget. It can be done. The Japanese have a very good way of doing that. Hopefully, we can learn some lessons from them. Primary drivers for the ABWR, at least from GEH's perspective, were we wanted to enhance the safety, and we wanted to improve the constructability and maintainability, as well. Some of the major design enhancements we made, we have an improved containment design. We went to pretty much a right cylinder design that's kind of a combination of our

Mark II and our Mark III pressure suppression containments.

It also is a reinforced concrete containment vessel with a steel leakage liner. It is a relatively compact reactor building. The emergency diesel generators are actually in the reactor building, so they moved a lot of equipment into a pretty small footprint.

We'll note that with the inclusion of the reactor internal pumps and the removal of our external recirculation loops, we were able to show, for all our design basis accidents, that we never have core uncovery.

So there's always water over the core, very little heat up when you do go into a transient situation. Although our reference design was based on the Japanese design at K6 and K7, our probabilistic risk assessment people led us to include some additional items in the design. Here's a list of several of those that we did. The reason I'm noting this is we do believe that these are -- if they hadn't been in the design, probably would have been added to address the post-Fukushima tsunami event. I'd like to point out that we were being proactive. We were looking at the design to identify some safety

enhancements. This is a list of those.

The first of those is we have an air-cooled combustion turbine generator that serves as an alternate AC power source. For the certification, the ABWR is classified as an alternate AC power plant for the station blackout rule. Having said that, we do have a reactor core isolation cooling pump, which also provides us with an AC independent means of cooling the core in the event of a station blackout.

We have what we call the AC independent water addition system, ACIWA. Very fancy way of saying we tied the fire water into several of our safety-related systems so that if we don't have other means of injecting water, we can use the fire water storage tanks and the fire water pumps to pump water into the reactor pressure vessel, into the containment, and into the spent fuel pool.

We also had a means of passively addressing cooling of core debris that would have melted through the bottom of the vessel in the event of a severe accident. This used thermally-actuated valves, kind of like what you have in the overhead here. What they did was they opened up and they allowed water from the suppression pool to float over to the debris that would have relocated down to the

bottom of the drywall.

Then finally, containment overpressure protection system, we did engineer in a leakage path or a vent path. That is from the suppression pool air space, so that we do credit scrubbing of any release. I believe we had a decontamination factor of ten that we credited for gasses and about 100 for particulate matter. That just allows that excess pressure to vent out to the atmosphere.

MEMBER REMPE: Is it a multi-unit application?

MR. BEARD: No.

MEMBER REMPE: It's just a single --

MR. BEARD: Following the EPRI guidance, it was designed as a single-unit standalone plant.

MEMBER REMPE: Okay.

MR. BEARD: So there's no sharing of systems.

MEMBER REMPE: I was going to ask about the standby gas treatment system, but I suppose, then, that doesn't come up in the application.

MR. BEARD: Quickly, the reactor core power of 3,926, that is kind of an edifice from Japan. They license on electrical output. The electrical output, when they backed it out, we got 3,926

megawatts of thermal. Eight hundred and seventy-two fuel bundles, very similar to our operating fleet, using the latest and greatest technology, 12 feet in length, 3.7 meters in length.

We characterize it as a moderate power density, 51 kilowatts per liter. To control that, we have 205 control blades. We did introduce -- one of the other major design enhancements with the ABWR is what we call our fine motion control rod drive. Prior to that, we actually had what we called a locking piston.

It was purely hydraulic insertion and withdrawal of the control blades, six-inch increments. With the fine motion control rod drives, we went to an electric motor that drives the lead screw. We now get five eighths of an inch increments for each notch position.

We also maintained the ability to hydraulically SCRAM. We actually have diverse means of inserting control blades. We can do it either hydraulically, which is the preferred safety-related means, or if that should fail for whatever means, we can also drive them in electrically.

MEMBER BLEY: How long does that take?

MR. BEARD: Hydraulic --

MEMBER BLEY: No, the --

MR. BEARD: To put this in perspective, the hydraulic SCRAM, less than two seconds, the electric drive in is a little less than two minutes from full out.

MEMBER MARCH-LEUBA: Since we are curious, on the control room, do you still use the CO248 display for control rod, or do you use inches?

MR. BEARD: I actually don't know. I have not been in the control room.

MEMBER MARCH-LEUBA: It's likely you used the CO248.

MR. BEARD: Yes. I'm sure we're going to get an answer. This is just an overall flow chart of the ABWR. I won't spend a lot of time talking about it, but given the power rating, a single high-pressure turbine, followed by three low-pressure turbines in series, so pretty standard conventional side of the plant. On the nuclear side, you see a pressure suppression containment denoted there, and then I'll talk about a little bit of the safety-related systems on the next slide. The approach for the ABWR was we had three divisions of safety related equipment.

They were operated by four divisions of instrumentation and control logic that was making the

decisions, but each of those three divisions has both a high-pressure injection capability, as well as a low-pressure injection capability.

One of those high-pressure injection capabilities is the reactor core isolation cooling turbine and pump that I spoke of, which provides us with an AC independent means. The three high-pressure systems are sufficient to inject enough cooling water to maintain the core cool in the event of -- should we have an isolation event.

Then the low-pressure systems inject plenty of water to handle all the break scenarios. As I said before, design basis accident point, we never have core uncovery. In the event that we go beyond design basis and we only have a single pump injecting, any one of our five motor-driven pumps is sufficient to keep enough water into the core to maintain adequate cooling.

CHAIRMAN RICCARDELLA: No recirculation, though, internal pumps.

MR. BEARD: Yes, reactor internal pumps. That's these yellow cans hanging down here. There are ten of them. They're about 700 kW each. Excuse me -- yes, 700 kW each. The next picture's just that of an artist's rendering of what this plant looks like. You

can see that the turbine structure is orientated perpendicular to the reactor building.

Somewhat interesting is the control building is located in between our reactor building and the turbine building here. The vast majority of the control building is actually below grade, including the main control room, itself. Timeline, I don't need to read this to you. It's there for your information.

It has been a relatively long and lengthy process. We've gotten through it, and we're glad to be at this final stage and hope to get good report out from the Committee today. In our original submittal to the NRC, as part of our renewal request, these are some of the items we addressed, aircraft impact, obviously a post-9/11 requirement. We did have some containment re-analysis we had to do based on some knowledge we gained from further projects. We did some selected design updates, and we also corrected a couple of errors that had been identified by GEH, again, in some of the construction projects we had going on. In addition to that, the NRC developed a list of 28 topics that they sent to us in a letter.

That list actually grew to 39 by the end of the renewal process, but we've worked our way

through all those. Anticipating the question, out of those 28 topics, there were several that GE proposed not be addressed. An example of that would be upgrading of the digital instrumentation control system.

As we mentioned last time we were here, because of the speed that technology continues to evolve, we felt that wasn't a worthwhile investment at this time because we did not know when our next potential customer would be and that we would upgrade at that time as part of the license application.

MEMBER REMPE: I apologize if this was brought up during the subcommittee meeting, but with the errors identified by GEH, did you look to understand why those errors occurred and provide yourself some sort of assurance that there wouldn't be other errors?

MR. BEARD: Yes. Following all the GE practices, we go through a corrective action program that we look at what are the root causes for these things. An example of that, that I talk about here later -- I'll go ahead and do it now -- is for the containment overpressure protection system, we did not have exact pipe routing at the time we were submitting this design.

We had some design assumptions on what the pressure losses would be in that system. When we finally got around to designing one of those for one of our projects, we determined that we were non-conservative with some of our design assumptions for that, so went back and corrected it, indicated what prior diameters were needed and how many elbows and flow restrictions we had in there in order to maintain the assumptions we had for the safety analysis.

MEMBER REMPE: You looked for any other possible situations similar to that and didn't find any.

MR. BEARD: Correct.

MEMBER REMPE: Sounds good. Thanks.

MR. BEARD: I keep forgetting I have to drive myself. Some of the significant design changes that we did incorporate listed here. Post-Fukushima 1, we did add two safety-related wide-range spent fuel level monitors. That gives a time to main reflectometry concept. We did enhance our ECCS suction strainers to address continuing concerns about plugging the strainers in the event of a LOCA event.

We had a new fuel vault in the original design, just a big hole in the operating floor on the

refuel floor, where we would put -- initial concept, when you brought the new fuel on site, inspected it, you would put it in there and store it. Most utilities, I think maybe all utilities, have stopped doing that.

They pull the fuel out; they inspect it; and they go ahead and put a channel on it and put it in the spent fuel pool, so it is ready for that, which eliminates a handling step and the potential to damage the fuel while doing that.

We did address the NRC Bulletin 2012-01 dealing with the out-of-phase current issues that were identified at -- I know it was an Exelon site in the Midwest, but I can't remember the exact site. We did some design changes to our electrical distribution system to monitor and to detect an out-of-phase condition, and then to isolate the out-of-phase condition and allow the diesel generators to come on and support the necessary safety functions.

MEMBER BLEY: That solution's in the design cert.

MR. BEARD: That solution is in the renewal design cert. I would point out that we did have commitments in the initial design that one of our three safety-related divisions was to actually be

powered from a different offsite source. We had requirements for two offsite sources.

We had a commitment that at least one of the three had to be on the auxiliary site. Fukushima Recommendation 4.2 mitigation strategies. We included several items to do that. We enhanced our fire protection system connections. We originally only had one external connection. We added a second external connection on a different face of the building, just to address the possibility that debris might have blocked access to the original thing. We did that.

MEMBER BLEY: A couple years ago, we had a presentation from NEI and the owners' group on the PWR strategies for using FLEX and other systems. Is that part of the design cert, or is that going to be -- the procedures for using all of that going to be done later?

MR. BEARD: Anything that was administrative or procedural in nature was deferred until we had an applicant for license.

MEMBER BLEY: Makes sense. Okay.

MR. BEARD: The change to the containment overprotection system I talked about already. Then we included some other changes to enhance the capability to implement the FLEX strategy, as outlined by NEI.

That completes our prepared comments. I'm prepared to answer any other questions.

MEMBER KIRCHNER: This is, perhaps, a late-in-the-game detailed question. Where are your FLEX connections to the fire mains and such?

MR. BEARD: Let me see.

MEMBER KIRCHNER: Are they between the control room and the reactor building? It's a leading question because I'm looking at your layout and I just noticed -- the control room's in the middle.

MR. BEARD: The control room's here, yes.

MEMBER KIRCHNER: It's in a hardened building, right?

MR. BEARD: It's in a reinforced concrete --

MEMBER KIRCHNER: But you've got your steam lines running right over, right?

MR. BEARD: Yes.

MEMBER KIRCHNER: And feedwater lines.

MR. BEARD: Mm-hmm.

MEMBER KIRCHNER: Are the FLEX connections into between those two buildings, or are they somewhere else?

MR. BEARD: We have two sets of FLEX connections. There's one over on what I call the

south wall.

MEMBER KIRCHNER: Okay, you answered my question.

MR. BEARD: Then there's another one over on this wall.

PARTICIPANT: Which wall?

MR. BEARD: This east wall. North is this side of the building right here, east, south, west.

PARTICIPANT: Thank you.

MR. BEARD: Mm-hm.

MEMBER MARCH-LEUBA: You mentioned there was a connection to add water for a very long-term --

MR. BEARD: Yes.

MEMBER MARCH-LEUBA: Where would that come from?

MR. BEARD: We'd use the fire water system. We have two 500,000-gallon tanks.

(Simultaneous speaking.)

MEMBER MARCH-LEUBA: So you just refill the fire tanks outside the containment, and then use that piping to come in?

MR. BEARD: We use the fire water system to connect to the residual heat removal system, and then the pipes within the residual heat removal system give us the capability to flow water either to the

containment, whether that be the dry well or the wet well, into the pressure vessel, itself, or into the spent fuel pool.

MEMBER BLEY: Did you build in any special filtering for that water or just --

MR. BEARD: No, we figured, at that point, that --

PARTICIPANT: Water's water.

MR. BEARD: Water's water. It's not salted water. It's clean water, but it's not --

MEMBER MARCH-LEUBA: You never know what.

MR. BEARD: It's not demineralized or anything like that. The answer back to the question about rod position indication, it's 0 to 200.

MEMBER MARCH-LEUBA: Two hundred.

PARTICIPANT: Two hundred steps.

MEMBER MARCH-LEUBA: You have to have some additional training for operators. I would have gone to 100 percent, maybe.

MR. BEARD: Any additional questions? If not, we thank you for your time and interest.

CHAIRMAN RICCARDELLA: Okay, with that, we'll bring up the staff for their presentation.

MR. SHEA: Good afternoon. My name is James Shea. I'm the staff project manager for the

ABWR DC renewal. Today, the staff will present an overview of the GEH design certification renewal review, and we'll review the ABWR DC renewal upcoming schedule activities and rulemaking.

As mentioned before, the ABWR is the only Generation 3 nuclear plant in operation today, not in the United States, at least yet, anyway. The ABWR was initially certified in 10 CFR Part 52, Appendix A, on May 12, 1997. The ABWR DC renewal application was submitted on December 7, 2010. In a July 20, 2012 letter, the NRC staff identified proposed DCD design changes that the staff believes should be considered for renewal. GEH provided, Revision 6 of the DCD of the ABWR, on February 19, 2016, in response to the staff-requested design changes, and then the staff completed its supplemental SER at the end of June of this year. GEH submitted the ABWR DC renewal application under Subpart B, standard design certifications of 10 CFR Part 52.

Scope of the ABWR DC renewal included a total of 39 design items proposed by staff or submitted by GEH. We talked about the 28 -- there was 28 specific staff items that were requested. Out of those, 22 were accepted by GEH, and six items were -- GEH wrote back to us that they thought they had

already covered. We re-reviewed those. Then in a February 2018 letter, we agreed with that assessment and wrote an additional assessment from the staff side in that letter.

MEMBER MARCH-LEUBA: How many of those were related to the aircraft impact analysis, just the main major analysis?

MR. SHEA: None of those. You mean of the 28 design items?

MEMBER MARCH-LEUBA: Mm-hm.

MR. SHEA: Actually, aircraft impact was submitted originally. That's one of the -- if you look at that 11 additional design items, it was actually submitted with the initial renewal.

MEMBER MARCH-LEUBA: Okay, so it wasn't additional.

MR. SHEA: As part of the rule of Part 52, it required AIA, if it wasn't already completed, which, in this case, prior to its original certification, the AIA rule had not been promulgated; therefore, as part of this rule, it is required to submit an AIA.

MEMBER MARCH-LEUBA: I understand from the subcommittee that was one of the main efforts.

MR. SHEA: That was, yes. It was one of

the main efforts. That's why I wanted -- my one particular staff member who worked on that, that was probably the most significant item that was addressed as part of the renewal.

Some of the other key significant renewal design changes included ECCS suction strainers, Fukushima design enhancements that GEH talked about. That included -- they talked about the AC independent water addition, and also connections to enable offsite sources to come in and connect to a fire truck or other water sources, in order to -- for a COL applicant, essentially, to meet the mitty bitty rule (phonetic), included AC connections offsite -- again, offsite, non-safety-related electrical generator could be brought onsite and connected to safety-related 1E electrical components, in order to mitigate a beyond design basis event, and EP enhancements mostly related to staffing, and also fuel pool instrumentation.

We just talked about two sets of fuel pool safety-related instrumentation, redundant. I think we mentioned it at the subcommittee meeting. They're backed by the AC and DC batteries that are backup, those two safety-related instruments. The ABWR AIA, mentioned that. That was submitted with the original renewal.

The PCT modification, that came up very late in the game, actually, probably toward the end of our review, when the staff noticed that PCT should have been updated for the renewal. Then GEH went back and did some changes based on 50.46 reporting requirements and made some adjustments to their evaluation model, and then resubmitted. Finally, a containment overpressure protection system, which they talked about, also, as being -- that came in, again, from GEH, originally, because they found an error in their analysis. Just as an example, we used the AIA as an example of the 39 items that we addressed, in this particular case, the SER Supplement, Chapter 19, Section 19.5, Aircraft Impact Assessment.

GEH submitted its assessment, again, initially, with the renewal. Changes included enhanced fire protection design features and ITAAC that ensures penetrations are not installed on the control building roof without an AIA cognizant engineer review.

In short, GE didn't change any parameters, as far as the design, the walls, the locations, the buildings, essentially took the NEI template for doing AIA and applied it and verified that the aircraft impact would not adversely impact the plant or meet

the parameters for ensuring safety of the plant.

Next, what we have is scheduled -- after this meeting, we do have our rulemaking that we'll -- we already had some preliminary rulemaking meetings. We'll kick that off, essentially, after this Phase 3 of the ACRS review. That schedule would include about a year to 14 months for rulemaking. That will go in parallel with the final SER that the staff will work on and have published for the rule effort, which would be Phase 4, FSER, with no open items. With that, I just want to summarize everything. The staff evaluated the GEH-proposed design updates to the ABWR and validated the findings in NUREG-1503 and NUREG-1503, Supplement 1.

This ABWR DC renewal safety evaluation report, Supplement 2 to the NUREG-1503, documents the NRC staff's review of GEH's application to renew the ABWR DC. Except as modified the supplement, the findings made in NUREG-1503 and its Supplement 1 remain in full effect.

The NRC staff made its safety determinations on specific modifications and amendments proposed by GEH as part of the DC renewal application. These modifications and amendments were found to meet the applicable regulatory requirements

and are therefore accepted. That ends the staff presentation, unless there's any questions.

CHAIRMAN RICCARDELLA: Thank you. Are there any --

MEMBER KIRCHNER: I'm sorry, go ahead.

CHAIRMAN RICCARDELLA: Are there any further questions from the Committee?

MEMBER KIRCHNER: Yes, Jim, just for the record, what is the expected time for the staff to complete a rulemaking on something like a DC renewal?

MR. SHEA: I think the goal that we have -- currently, we are planning to have a direct final rule, so it won't go out for the normal rulemaking process. That should shorten the process. What we have in planning phases, right now, is expected about 12 to 14 months.

MEMBER KIRCHNER: Why does it take so long? It's not a safety question; it's a process question.

MR. SHEA: Part of the fact that we still have Phase 4, the FSER. Step 1 is GEH will submit us their Revision 7 to the DCD, which they haven't done yet. They're still going through their final validation process on that. Once they submit that to us, we use that as the basis to go back and look at

all the SERs that we completed that were based on markups of Rev. 6.

For example, AIA, we had some RAIs following the Revision 6 to the DCD that were then addressed and included markups, which will now be incorporated into Revision 7. At that point, the staff -- we will verify all those changes were made in Revision 7. That's going to take some time, probably until the end of the year. Right now, the schedule has us completing the FSER in -- I think we have it in March 2020.

MEMBER KIRCHNER: Yes, that's what you have on Slide 9.

MR. SHEA: That all will depend on when we get the DCD. It should only take us -- in this particular case, it should only take us a few months to complete the FSER process. You need the FSER and the final DCD to actually submit it as a rule. We've got to have those two main administrative pieces done before we can actually go to final rule.

MEMBER KIRCHNER: I should have framed my question differently. When you get done with the final SER with no open items in March of next year, which is your target, how much time, then, is allocated for the rulemaking, itself?

MR. SHEA: Like I said, that can be done -- we should be kicking the rule off within a month or so. That would take us, from that point, 12 to 14 months. That includes, in parallel, us completing the FSER. That's part of that process. It wouldn't be in addition.

MEMBER KIRCHNER: Thank you.

MR. SHEA: Like I said, if we can get the DCD back to us even before -- I think we're scheduled to get it before the end of this year, the final DCD, we can then validate. Then this process could take shorter than our goal. That's what we would strive to do.

CHAIRMAN RICCARDELLA: Any other questions from members? Let's check for members of the public. Is there anybody in the room from the public who would like to make a comment on this?

Not seeing any, we will open the phone line and accept questions from anybody who happens to be on the line. Are there any members of the public out there who would like to make a comment on the ABWR design certification renewal? If so, please state your name and make your comments.

Not hearing any, we'll proceed. I think -- we're finished with the presentations. We do have

a draft letter, which --

MR. NGUYEN: Chairman, I guess I recommend a break, so I can make copies while we have the licensee here.

CHAIRMAN RICCARDELLA: Okay, we'll take a break until 2:00 p.m., 15 minutes, and then we will -- okay, we're off the record until the next topic, which is at 2:30.

(Whereupon, the above-entitled matter went off the record at 1:43 p.m. and resumed at 2:30 p.m.)

CHAIRMAN RICCARDELLA: Next topic on the agenda is Framatome's topical report on RAMONA5 for anticipated transient without SCRAM. A portion of this meeting will be open, and then we will close the meeting for a closed session. With that, I'll ask, Jane, do you want to make a comment?

MS. MARSHALL: Yes, I'll make a couple of quick opening remarks. I know the ACRS subcommittee had an opportunity to review this a month or so ago. Framatome has generalized this methodology, so it can now be used at -- they generalized it to a form that can be used at any BWR currently operating in the U.S. The methodology is directly applicable to Brunswick ATRIUM-11.

That's scheduled to be presented to the

ACRS subcommittee in November, since Brunswick proposed to use an identical analysis. The NRC staff, our contractor from Oak Ridge National Lab, and Framatome staff have demonstrated a -- have been very responsive to each other's needs. We've had an efficient and, we think, satisfactory completion of this complex review, without any challenges or delays. Thanks.

MEMBER MARCH-LEUBA: Framatome, you want to do some introductory remarks on the open session? Don't use any proprietary slides until we close.

MR. TINKLER: I guess we'll start with some introductions, here. My name is Dan Tinkler. I've been with Framatome, now, for 17 years. Fourteen of that has been working various stability methodology development projects, starting with our long-term stability solution methodology, RAMONA5 based, going up through various plant specifics, and now here, today, moving into the generic ATWS-I.

With me is Dr. Farawila. He has many decades of stability experience working various items, such as testing, methods development, pretty much spanning the gamut when it comes to the stability field.

MEMBER MARCH-LEUBA: Do you want to say

something on the topic of representation in open session?

MR. TINKLER: The generic ATWS-I methodology we're really going to present today is kind of -- it's not a new methodology. It's kind of the culmination of some plant-specific work and previous work we've done on the long-term stability solution. That's really kind of what we're going to present to you today.

MEMBER MARCH-LEUBA: Thank you. Mr. Chairman, at this point, I propose that we close the open session.

CHAIRMAN RICCARDELLA: Okay, the open session is closed. We'll now go into closed session.

MEMBER KIRCHNER: Jose, are we going to go closed, and then open again?

MEMBER MARCH-LEUBA: No, this is --

MEMBER KIRCHNER: Then something's missing, I would suggest. The public needs to know what's the answer. What I mean by that is has the staff review approved the topical? That should be part of the record.

MEMBER MARCH-LEUBA: It should. Let me give a summary. The staff has issued an SER.

MEMBER REMPE: Just slow down. First of

all, are we still in the open session, or did we close it? I thought I heard the gavel bang. Are we in the open session -- the transcriber's going to have to answer this -- or are we in the closed session?

COURT REPORTER: We're still in open.

MEMBER REMPE: In addition to this, we need to allow for public comments. That's good. Sorry to interrupt.

MEMBER KIRCHNER: The public can't really comment until they know what the answer is. We need a summary of what was presented and what was approved.

MEMBER MARCH-LEUBA: The staff has reviewed the proposed methodology to calculate ATWS-I transients in BWRs, on a generic basis, and found it acceptable. That's what their SER says.

CHAIRMAN RICCARDELLA: With that, before we go into the closed session, we'll ask are there any members of the public in the room that would like to make a comment?

(No response.)

CHAIRMAN RICCARDELLA: Are there any members of the public on the phone line? Which I guess is open now, because Ashley's on the line. Ashley, are you there?

MS. SMITH: Yes, I'm still here.

CHAIRMAN RICCARDELLA: Is there anybody else on the line, besides Ashley?

Hearing none, I assume there's nobody from the public who would like to make a comment. Is it okay for Ashley to be on the open line? What if someone else --

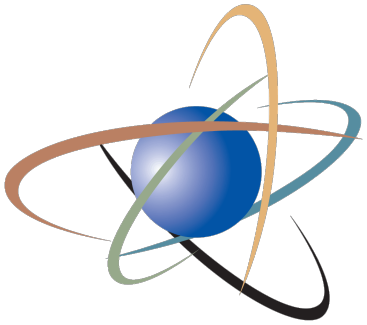
MEMBER MARCH-LEUBA: This is the closed line, right?

MS. MARSHALL: She'll use the closed line. She has that number now, but she's not on it yet.

MS. SMITH: Yes, thank you. I'll call back in.

CHAIRMAN RICCARDELLA: All right, very good, thank you. With that, we can close the open phone line, the public line.

(Whereupon, the above-entitled matter went off the record at 2:37 p.m.)



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UNITED STATES NUCLEAR REGULATORY COMMISSION

Protecting People and the Environment

Presentation to the ACRS Full Committee

Staff Safety Review of ABWR DC Renewal

October 2, 2019

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UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

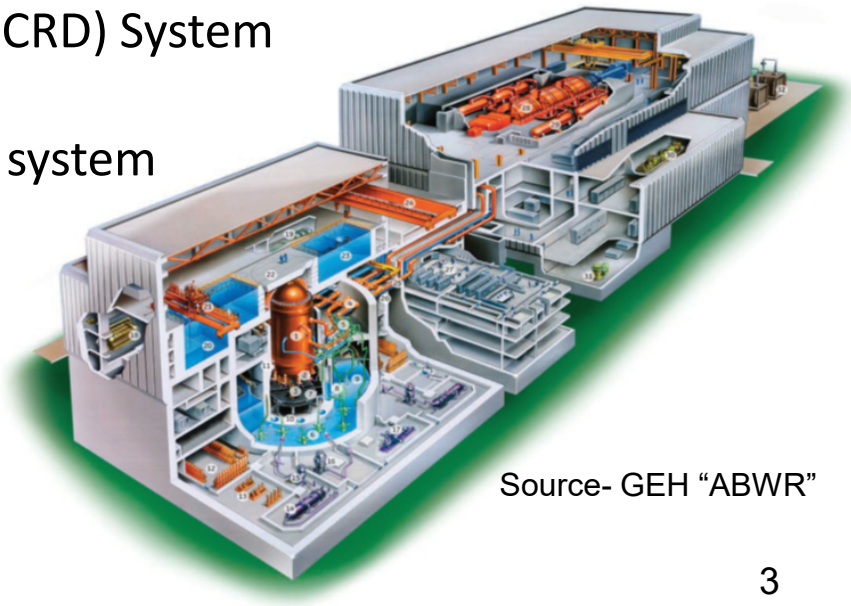
ABWR DC RENEWAL

Agenda

- Overview of the General Electric Hitachi (GEH) Advanced Boiling Water Reactor (ABWR) Design, Certification and Renewal.
- ABWR Design Certification (DC) Renewal Application
  - Regulatory Basis for DC Renewal
  - Design Change Items Proposed and Reviewed
  - Key Significant Design Changes
  - Staff Conclusions
- Schedule for the ABWR DC Renewal Activities

Overview of the ABWR Design

- Generation III Reactor with enhanced safety features
- ABWR is a single-cycle, forced-circulation, boiling-water reactor (BWR), with a rated power of 3926 MWt
- Reactor recirculation system applying internal pumps
- Advanced Fine Motion Control Rod Drive (CRD) System
- Main Control Room (MCR) with full digital system
- Reinforced concrete containment vessel



Source- GEH "ABWR"

ABWR DC Renewal Application Summary

- **May 1997:** Staff FSER NUREG-1503 Supplement 1 based on ABWR design control document (DCD) Revision 4.
- **May 12, 1997:** Initial ABWR DC Rule (Appendix A to Title 10, Part 52)
- **December 7, 2010:** GEH ABWR DC Renewal Application DCD Revision 5
- **July 20, 2012:** NRC staff Identified proposed changes including Fukushima Near Term Task Force Recommendations (NTTF) from SECY-12-0025
- **February 19, 2016:** GEH provided ABWR DCD Revision 6 in response to staff requested changes with GEH responses to those requests
- **June 28, 2019:** NRC staff completed Advanced Supplemental SER with no open items

DC Renewal Regulatory Basis

- Regulatory Requirements for DC Renewal Applications
  - 10 CFR 52.57, Application for renewal
  - 10 CFR 52.59, Criteria for renewal

- GEH submitted the ABWR DC renewal application under Subpart B, "Standard Design Certifications," of 10 CFR Part 52
  - Application included the ABWR DCD and an environmental report (ER).

ABWR DC Renewal Design Items

- **28 Design Items** Proposed by the staff for Consideration:
  - GEH accepted the changes proposed by the staff for 22 items and included the changes in the February 2016 DCD Revision 6.
  - 6 items not incorporated in revised ABWR DCD.

- **11 additional design items** identified at time of Renewal or during the review of the application.

- **39 Total Design Items Reviewed and Approved** in Supplemental SERs to NUREG-1503 or closed by letter.

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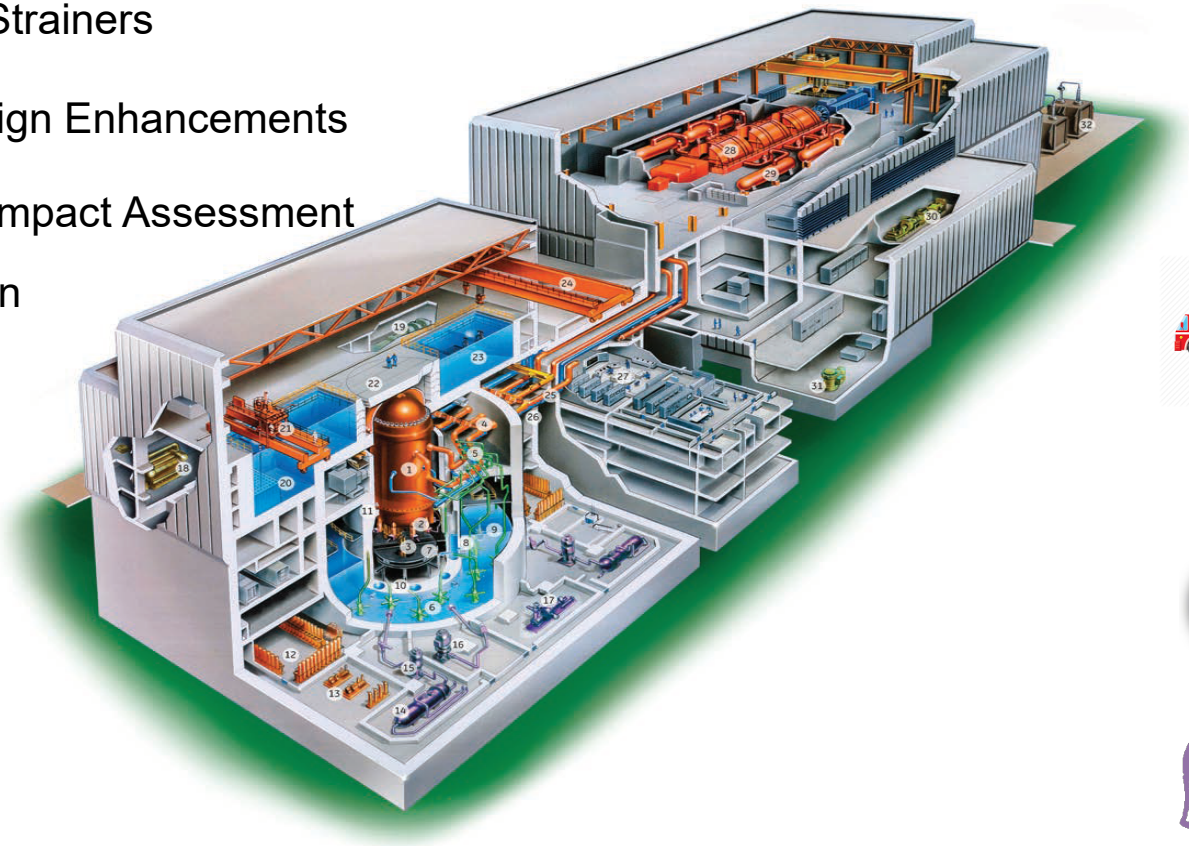
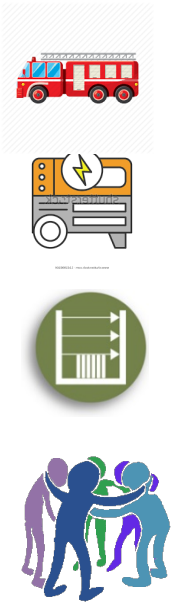
ABWR DC RENEWAL

Key Significant Renewal Design Changes

- ECCS Suction Strainers
- Fukushima Design Enhancements
- ABWR Aircraft Impact Assessment
- PCT Modification
- COPS





Source- GEH “ABWR”

ABWR DC Renewal

- **Issue 29 -AIA Aircraft Impact Assessment:**
- **Design Change Type - Modification**

SER Supplement Chapter 19 Section 19.5 Aircraft Impact Assessment**:**

Submitted as part of the DC Renewal (DCD Revision 5) - ABWR DCD Tier 2, Section 19G,Revision 6, GEH “Aircraft Impact Assessment," and proposed changes to Revision 6 of the ABWR DCD.

- ✓ Enhanced Fire Protection Design Features.

- ✓ Control Building (C/B) penetrations are not installed on the C/B roof without an AIA cognizant engineer review.

The NRC staff also finds that the applicant adequately described the key design features and functional capabilities identified and credited to meet 10 CFR 50.150(b), including how the key design features meet the acceptance criteria in 10 CFR 50.150(a)(1).

Schedule – Letter Dated 5/31/19

Key Milestones	Completion Date Actual - A Target - T
Application	
Received Design Certification Renewal Application	12/07/10 - A
Acceptance Review	
NRC to issue Acceptance Review Determination Letter	02/14/11 - A
Safety Review	
Phase 1 - Preliminary Supplemental Safety Evaluation Report (SER) and Requests for Additional Information	01/21/19 - A
Phase 2 - Advanced Supplemental SER with No Open Items	06/28/19 - A
Phase 3 - ACRS Review of SER with No Open Items	10/19 - T
Phase 4 - Final SER with No Open Items	03/20 - T
Rulemaking	
Issue final rule	TBD

ABWR DC Renewal NRC Staff Conclusions

- The NRC staff evaluated the GEH proposed design updates to the ABWR and validated the findings in NUREG–1503 and NUREG–1503 supplement 1.

- This ABWR DC Renewal Safety Evaluation report, Supplement 2 to NUREG–1503, documents the NRC staff's review of GEH's application to renew the ABWR DC. Except as modified by this Supplement, the findings made in NUREG-1503 and its Supplement 1 remain in full effect.

- The NRC staff made safety determinations on the specific Modifications and Amendments proposed by GEH as part of its DC Renewal Application.

- These Modifications and Amendments were found to meet the applicable regulatory requirements and are therefore acceptable.

- Thank You!

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Backup Slides

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ABWR DC Renewal

List of Abbreviations Used

- ABWR – Advanced Boiling Water Reactor
- ac – Alternating Current
- ACS – Atmospheric Control System
- ACRS – Advisory Committee on Reactor Safeguards
- ACIWA - Alternating Current (ac) Independent Water Addition System
- AIA – Aircraft Impact Assessment
- ATWS – Anticipated Transient Without Scram
- BWR – Boiling Water Reactor
- C/B – Control Building
- COL – Combined License
- COPS- Containment Overpressure Protection System
- CRD- Control Rod Drive
- DBA – Design Basis Accident
- DC – Design Certification
- DCD – Design Control Document
- ECCS – Emergency Core Cooling Systems
- EP – Emergency Planning
- ER – Environmental Report
- GEH- General Electric Hitachi
- I&C – Instrument and Control
- IEEE – Institute of Electrical and Electronics Engineers
- ITAAC - Inspections, Tests, Analyses, and Acceptance Criteria
- MBDBE- Mitigation of Beyond Design Basis Events
- MCR – Main Control Room
- NPSH – Net Positive Suction Head
- NTTF - Fukushima Near Term Task Force Recommendations
- NRC – US Nuclear Regulatory Commission
- RAI – Request for Additional Information
- RB – Reactor Building
- RG – Regulatory Guide
- RHR – Residual Heat Removal System
- RSP – Remote Shutdown Panel
- SER – Safety Evaluation Report
- SFP – Spent Fuel Pool
- SR – Safety Related
- SRP – Standard Review Plan
- SSC – Structure, Systems, and Components
- TS – Technical Specifications
- TSC – Technical Support Center

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ABWR DC RENEWAL

Item No.	Description	Type
1	**SER Supplement Chapter 2.0 Section 2.5 Geological, Seismological and Geotechnical Engineering**	**Modification**
2	**SER Supplement Chapter 2.3 Section 2.3.1, Regional climatology**	**Modification**
2	**SER Supplement Chapter 3 Section 3.3, Wind and Tornado Loadings**	**Modification**
2	**SER Supplement Chapter 3 Section 3.5.1.4.1 Missiles Generated by Natural Phenomena**	**Modification**
3	**SER Supplement Chapter 2.0 Section 2.6.8 ABWR Site Acceptability**	**Modification**
4	**SER Supplement Chapter 2.0 Section 2.6.2 Water Level (Flood) Design Site Parameters**	**Modification**
5	SER Supplement Chapter 12 Section 12.3 Radiation Protection Design Features	**Amendment**
6	SER Supplement Chapter 12 Section 12.2 Radiation Sources (SER covers Issues 6&7)	**Modification**
7	SER Supplement Chapter 12 Section 12.2 Radiation Sources (SER covers Issues 6&7)	**Modification**
8	SER Supplement Chapter 11 Section 11.4 Solid Waste Management System	**Modification**
9	SER Supplement Chapter 6 Section 6.2.1.9 Containment Debris Protection for ECCS Strainers	**Amendment**
10	SER Supplement Chapter 5.0 Section 5.4.8 Reactor Water Cleanup System.	**Amendment**
11	SER Supplement Chapter 9 Section 9.5.1 Fire Protection System	**Modification**

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ABWR DC RENEWAL

Item No.	Description	Type
12	SER Supplement Chapter 5.0 Section 5.2.5 Reactor Coolant Pressure Boundary Leakage Detection.	Amendment
13	SER Supplement Chapter 9.0 Section 9.1.1 New Fuel Storage	Amendment
13	SER Supplement Chapter 9.0 Section 9.1.4 Light Load Handling System (Related to Refueling)	Amendment
13	SER Supplement Chapter 9.0 Section 9.1.5 Overhead Heavy Load Handling Systems	Amendment
14	Update the Level 1 and 2 full-power probabilistic risk assessment (PRA) for the ABWR, including its description and results in Chapter 19 of the DCD.	Issue Closed
15	Complete a Level 1 and 2 shutdown PRA for the ABWR, including its description and results in Chapter 19 of the DCD.	Issue Closed
16	Update Appendix 19K to develop a comprehensive list of risk-significant SSCs.	Issue Closed
17	SER Supplement Chapter 13 Section 13.5 Plant Procedures	Amendment
18a	SER Supplement Chapter 4 Section 4.2 Fuel System Design	Modification
18b	SER Supplement Chapter 9 Section 9.1.2.1 Fuel Racks	Modification
19	SER Supplement Chapter 9 Section 9.1.2 New and Spent Fuel Storage (SER covers Issues 19&20)	Modification
20	SER Supplement Chapter 9 Section 9.1.2 New and Spent Fuel Storage (SER covers Issues 19&20)	Modification

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ABWR DC RENEWAL

Item No.	Description	Type
21	Replace obsolete (I&C) and data communication technology. The replacement design should conform to current instrumentation and control related regulations, industry standards, and regulatory guidance.	Issue Closed
22	SER Supplement Chapter 7.0 Section 7.7.1.2.1 Control Rod Ganged Withdrawal Sequence Restrictions	Modification
23	SER Supplement Chapter 3.0 Section 3.7.3, Seismic Subsystem Analysis	Modification
24	Apply the guidance from Regulatory Issue Summary 2008-05, Revision 1, to the existing ITAAC and submit revised ITAAC.	Issue Closed
25	Provide a control room design that reflects state-of-the-art human factor principles in accordance with 10 CFR 50.34(f)(2)(iii).	Issue Closed
26	SER Supplement Chapter 22 Sections 5.4.7 RHR, 5.4.7.1.1.10 ACIWA, 7.4.1.4.4 RSP, 8.3.4.4 1E Buses Chapter 16 TS	Amendment
27	SER Supplement Chapter 22 Sections 3.2.3 Safety Classifications, 7.5.2.1 Post Accident Monitoring System, 9.1.3 Fuel Pool Cooling	Amendment
28	SER Supplement Chapter 13 Section 13.3 Emergency Planning (SER Covers Issue 28&31)	Modification
29	SER Supplement Chapter 19 Section 19.5 Aircraft Impact Assessment	Modification
30	SER Supplement Chapter 6 Section 6.2.1.3 Short-Term Pressure Response	Amendment

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ABWR DC RENEWAL

Item No.	Description	Type
31	SER Supplement Chapter 13 Section 13.3 Emergency Planning (SER Covers Issue 28 & 31)	Modification
32	SER Supplement Chapter 19 Section 19.2.3.3.4 ABWR Containment Vent Design	Modification
33	SER Supplement Chapter 8 Section 8.2.5 NRC Bulletin 2012-01 Design Vulnerability	Modification
34	SER Supplement Chapter 6 Section 6.2.1.6 Suppression Pool Dynamic Loads	Modification
35	SER Supplement Chapter 14 Section 14.3.2.3.6 Structural Task Group Review	Modification
36	SER Supplement Chapter 1 Operating Experience Review (Chapter 1 SER Covers Issues 36 &37)	N/A
37	SER Supplement Chapter 1 Alternate Vendor/Changes to Chapter 1 SE (Chapter 1 SER Covers Issues 36 &37)	N/A
38	**SER Supplement Chapter 6 Section 6.3 Emergency Core Cooling Systems**	Modification
39	Supplement Chapter 19 PRA to discuss effect of design changes on PRA.	N/A

GE Hitachi Nuclear Energy

ABWR Design Certification Renewal

ACRS Committee meeting 2 October 2019

HITACHI

GEH Presentation

- ABWR Overview
- U.S. Design Certification Renewal Timeline
- Renewal Scope
- Significant Design Changes

HITACHI

ABWR Overview

- GEH's first ABWR began commercial operation at Kashiwazaki-Kariwa (K/K) in Japan, in 1996.
- Three additional ABWRs operational in Japan
- Two more under construction in Japan, and two in Taiwan.
- The ABWR is licensed in Japan and Taiwan, certified in the U.S., and approved in the UK (GDA)

HITACHI

ABWR Overview (cont.)

The ABWR was developed as a collaborative effort between GE, TEPCO, Hitachi and Toshiba

- First Plants were built at the K/K site as units 6 and 7

Primary Drivers were enhanced safety and improved constructability and maintainability

- Improved Primary Containment design
  - Combines features of the Mark II and III containments
  - Reinforced Concrete Containment Vessel (RCCV) with steel leakage liner
- Compact Reactor Building of primarily reinforced concrete
- No Core Uncovery during a Design Basis Accident (DBA)
- Reactor Internal Pumps (RIPs)

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ABWR Overview (cont.)

The U.S. NRC certified design incorporated additional features:

- Combustion Turbine Generator as an Alternate AC power source (air-cooled)
- AC Independent Water Addition (ACIWA) System using Fire Protection as diverse water source
- Lower Drywell Flooder utilizing passive thermally activated valves to flood the Lower Drywell in the event of an ex-vessel core melt
- Containment Overpressure Protection System (COPS)
  - Passive rupture disc venting from Suppression Pool Airspace

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ABWR Overview (cont.)
Reactor Specification

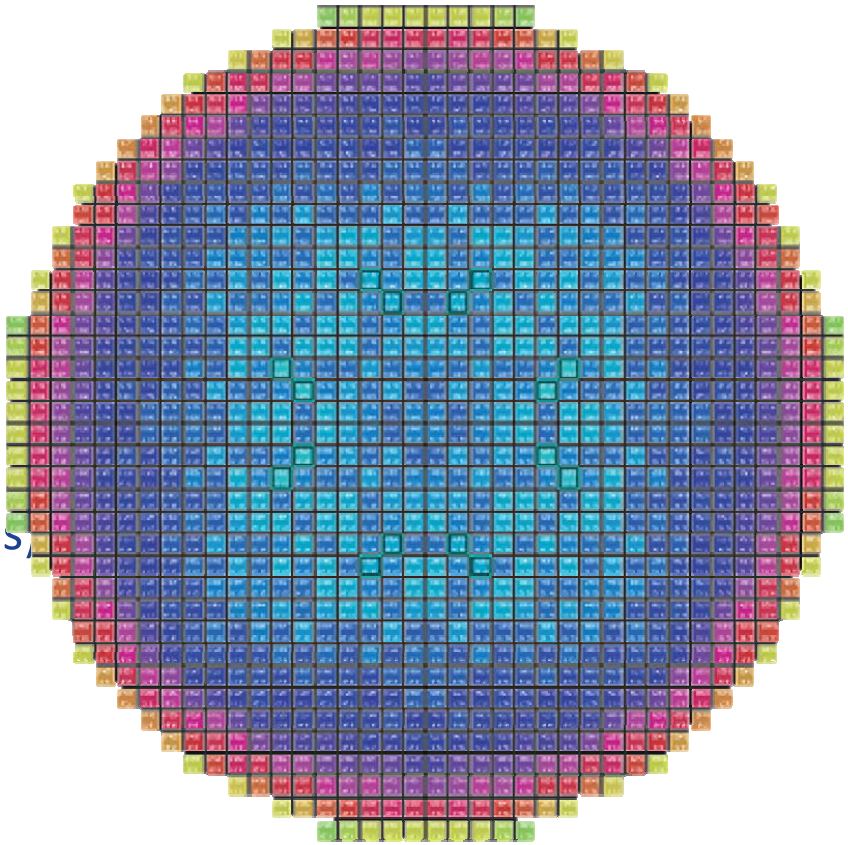
3926 Rated MWt

872 Fuel Bundles

- N- Lattice (symmetric water gap)
- Active Fuel Length (3.66 m; 12 ft)
- Moderate Power Density (51 kw/liter)

205 Control Blades

- Fine Motion Control Rod Drives (FMCRDs)
  - Reduced Fuel Duty
  - Fast Hydraulic Scram



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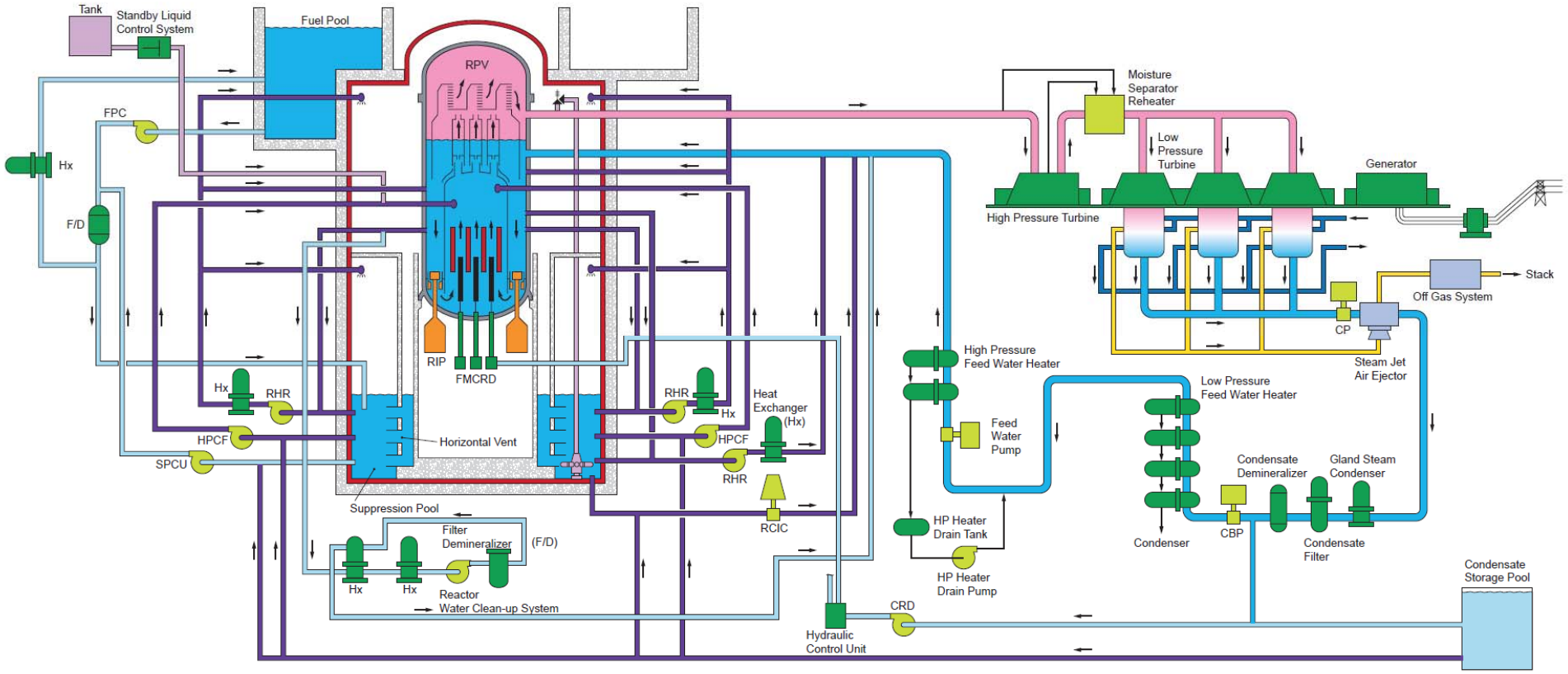
ABWR Overview (cont.)
Overall Flow Chart



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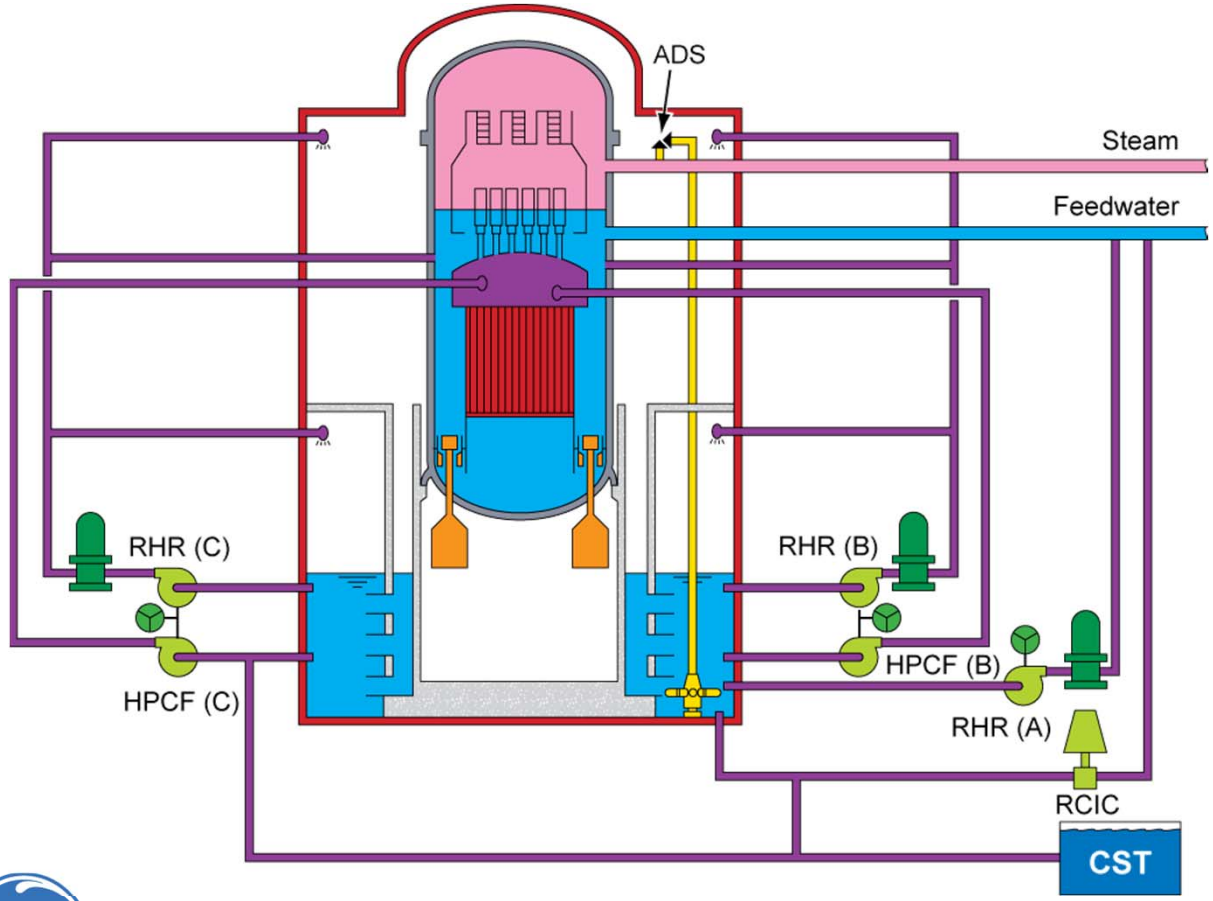
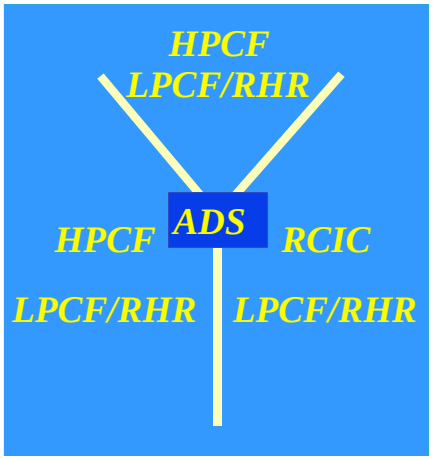
ABWR Overview (cont.)
Emergency Core Cooling Systems





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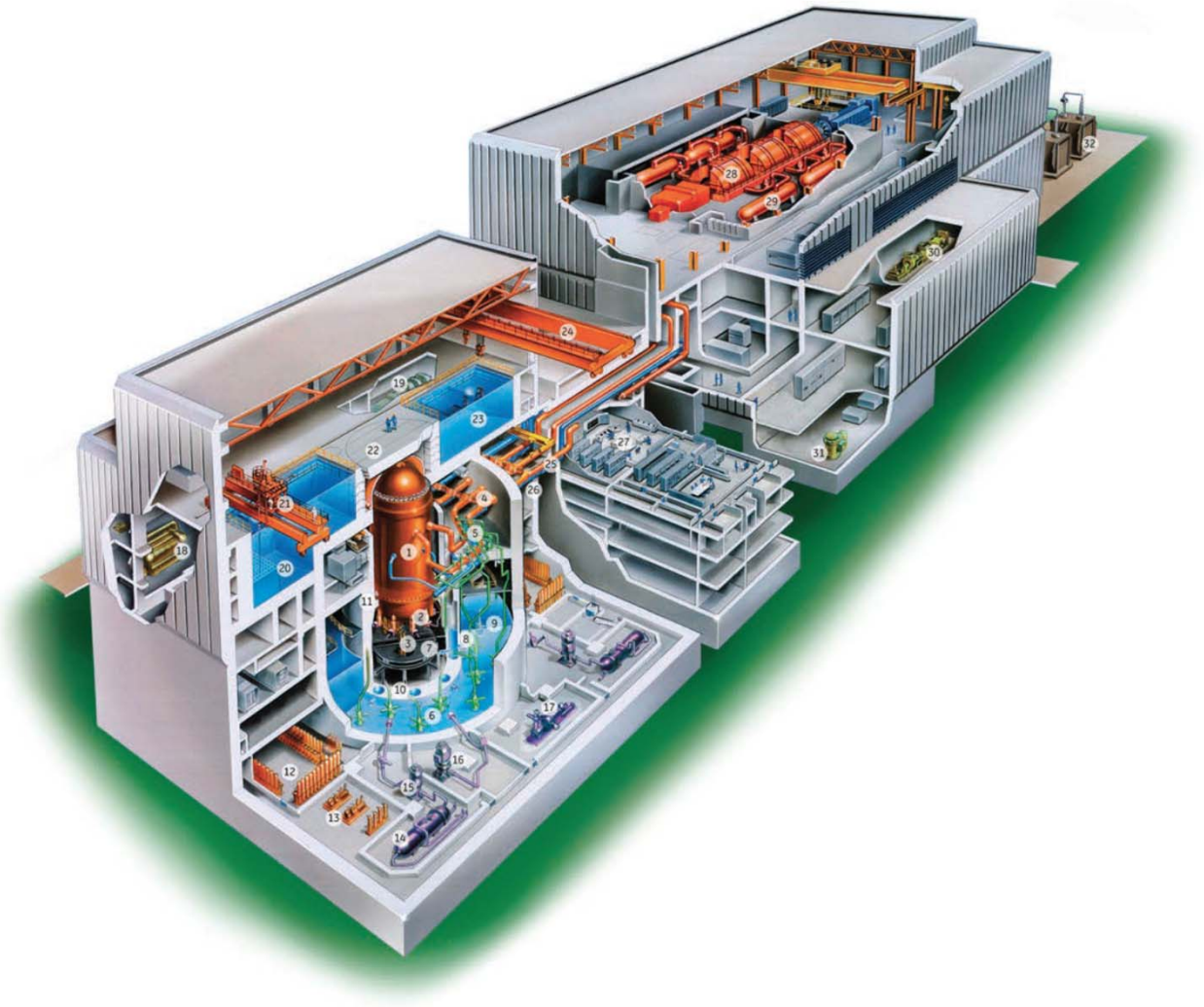
ABWR Overview (cont.)



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U.S. Design Certification Renewal Timeline

- Renewal Application Submitted (ABWR DCD rev 5) — Dec 2010
- Application Docketed by NRC — Feb 2011
- Initial Application Review Meeting — Mar 2011
- NRC Letter – Proposed Changes (28 items) — Jul 2012
- GEH response to NRC Letter — Sep 2012
- ABWR DCD revision 6 — Feb 2016
- Final GEH response (PCT) — Jan 2019
- ACRS ABWR Subcommittee meeting — Aug 2019

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Renewal Scope

Original Submittal

- Aircraft Impact Assessment
- Containment Re-analysis
- Selected design updates
- Corrected errors identified by GEH

NRC identified

- NRC originally identified 28 topics
- Final list was 39 items

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Significant Design Changes

- ABWR added two safety-related wide range spent fuel pool level
- Enhanced ECCS Suction Strainers
- Deletion of new fuel vault
- Addressed NRC bulletin 2012-01
- ABWR DCD Fukushima Recommendation 4.2 - Mitigation Strategies
- ABWR DCD COPS Size Corrections
- Included changes needed to enhance FLEX

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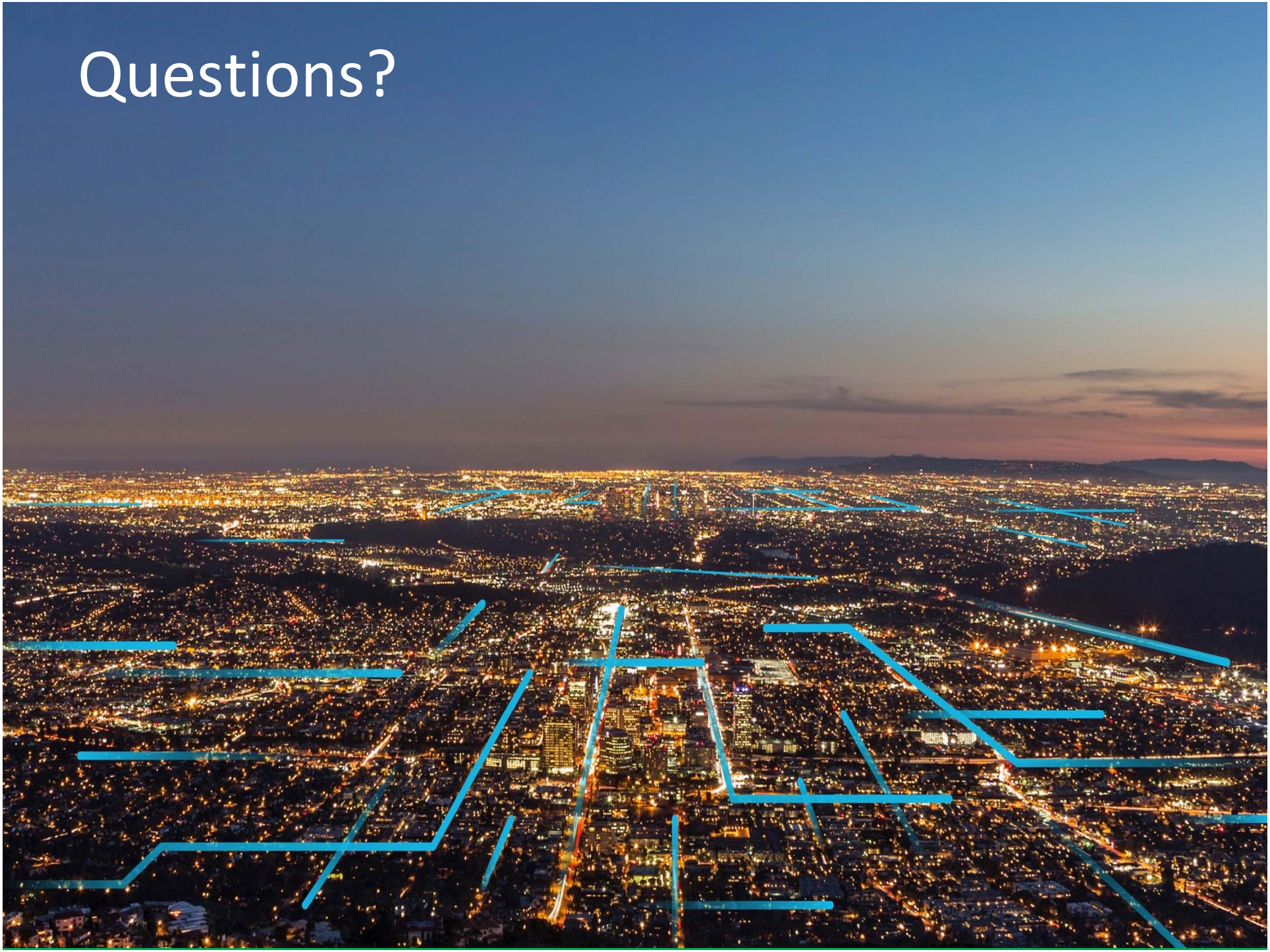
Questions?

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