Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title:	Advisory Committee on Reactor Safeguards BWRX-300 Design-Centered Subcommittee
Docket Number:	(n/a)
Location:	videoconference
Date:	Wednesday, August 20, 2025

Work Order No.: NRC-0433 Pages 1-137

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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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 DESIGN-CENTERED REVIEW SUBCOMMITTEE

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WEDNESDAY

AUGUST 20, 2025

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The Subcommittee met via Videoconference, at 8:30 a.m. EDT, Craig Harrington, Chair, presiding.

SUBCOMMITTEE MEMBERS:

CRAIG D. HARRINGTON, Chair

VESNA B. DIMITRIJEVIC

GREGORY H. HALNON

WALTER L. KIRCHNER

ROBERT P. MARTIN

THOMAS E. ROBERTS

ACRS CONSULTANTS:

RONALD G. BALLINGER

DENNIS BLEY

DESIGNATED FEDERAL OFFICIAL:

QUYNH NGUYEN

ALSO PRESENT:

KELLI BANKS, GE Vernova

DAVID HINDS, GE Vernova

SCOTT HUNNEWELL, Tennessee Valley Authority

BRIAN McDERMOTT, Tennessee Valley Authority

RAY SCHIELE, Tennessee Valley Authority

P-R-O-C-E-E-D-I-N-G-S

8:30 a.m.

CHAIR HARRINGTON: The meeting will now come to order.

I'm Craig Harrington, Chairman of the BWRX-300 Design-Centered Subcommittee.

We've got feedback -- okay.

ACRS members in attendance in person, Greg Halnon, Robert Martin, and Thomas Roberts. ACRS members Vesna Dimitrijevic, Walt Kirchner -- and Walt Kirchner are participating virtually via Teams. We, I think, have consultants Ron Ballinger and Dennis Bley participating virtually.

Have I missed anyone, either ACRS members or consultants, please speak up now?

MEMBER DIMITRIJEVIC: Vesna is also here. I just joined. Good morning.

CHAIR HARRINGTON: Okay, very good. Thanks, Vesna.

Quynh Nguyen of the ACRS staff is the Designated Federal Officer for today's meeting.

No member conflicts of interest were identified for today's meeting. And we have a quorum.

The ACRS was established by statute and is governed by the Federal Advisory Committee Act, or

FACA.

NRC implements FACA in accordance with our regulations. Per these regulations and the Committee’s bylaws, this ACRS speaks only through its published letter reports.

All member comments should be regarded as only the individual opinion of that member, not the Committee position.

All relevant information related to ACRS activities such as letters, rules for meeting participation, and transcripts are located on the NRC public website and can be easily found by typing about the ACRS in the search field on NRC’s home page.

The ACRS, consistent with the Agency’s value of public transparency and regulation of nuclear facilities provides opportunity for public input and comment during our proceedings.

For this subcommittee meeting, we have received no written comments. Written statements may be forwarded to today’s Designated Federal Officer.

We have also set aside time at the end of this meeting for public comments which will be added to the record and considered by the Committee in future deliberations.

However, the Committee does not plan to --

on responding to specific comments during today's meeting.

The purpose of today's meeting is to hear a general overview from TVA of their construction permit application for the Clinch River site.

Following that overview, hear a general presentation of the BWRX-300 Design from GE Vernova, which is the designated design for this site.

Previously, in December 2019, an early site permit, ESP, was issued for the Clinch River site.

A transcript of the meeting is being kept and will be posted on our website.

When addressing the committee, the participants should first identify themselves and speak with sufficient clarity and volume so that they may be readily heard. If you are not speaking, please mute your computer on Teams.

If you are participating by phone, press star six to mute your phone and star five to raise your hand in Teams. The Teams chat feature will not be available for use during the meeting.

For everyone in the room, please put all your electronic devices in silent mode and mute your laptop microphone speakers. In addition, please keep sidebar discussions in the room to a minimum since the ceiling microphones are live.

For the presenters, table microphones are fairly unidirectional and need to be close. So, you need to speak into the front of the microphone to be heard online.

And finally, if you have any feedback for the ACRS about today's meetings, we encourage you to fill out the public meeting feedback form on the NRC's website.

And, with that, Scott Hunnewell from TVA will kick off the meeting.

MR. HUNNEWELL: Thank you. I'm Scott Hunnewell. I'm the Vice President of the New Nuclear Program for the Tennessee Valley Authority.

Good morning, and thank you for inviting us to share our work on the BWRX-300 at the Clinch River site in Oak Ridge, Tennessee with you today.

So, TVA was formed in 1933 as part of the New Deal by FDR. In the 1940s, we primarily deployed dams, hydroelectricity. 1950s, fossil fuel. The 1960s saw us enter the nuclear realm. And then, 1970s, primarily natural gas generation.

And as we look forward to the future, we look at a diverse slate of generation, including, potentially, new nuclear power.

So, East Millinocket, Maine, is a small town about an hour north of Bangor on the edge of the Allagash Wilderness.

I want to read portions of the valedictorian speech from the 1958 high school graduating class.

The Italian navigator has arrived in the New World. With these humorously cryptic words, Arthur H. Compton telephoned James B. Conant to announce the dawn of the Atomic Age. The navigator was Enrico Fermi, the Italian-born American physicist.

The date was December 2nd, 1942. The scene, a squash court under the stadium of the University of Chicago. There, a select audience watched, not sure what to expect, as a cadmium rod was pulled slowly, foot by foot, from a strange structure of carbon and uranium that looked somewhat like a giant beehive.

At 3:20 p.m., Fermi achieved the first self-sustaining atomic chain reaction and operated the world’s first atomic furnace. He had, indeed, arrived in the New World.

The achievements of atomic energy are possible because of the atomic furnace, more accurately called a nuclear reactor.

The reactor is a very unusual machine, for it does two separate and entirely different jobs. Both its abilities are of great practical value and they are being developed simultaneously.

On the one hand, the reactor is a furnace. It makes heat just as a furnace that consumes coal, oil, gas, wood, gasoline, or any other fuel does.

The atomic furnace, flameless, strangely quiet, almost entirely automatic, is used to make steam for generating electricity. Generating electricity is probably the biggest use for atomic heat, but there are other important applications.

It can propel ships, and an atomic engine now keeps vessels cruising smoothly over vast distances without need for refuel.

An unusual type of atomic furnace could drive a railroad locomotive and still others are developed to fly airplanes and rocket ships.

Medicine and agriculture have made perhaps the biggest use of atomic energy so far. Radioactive tools have given us new plants, fertilizers, and insecticides, as well as improved cancer treatments.

Despite the Manhattan Project's preoccupation with war work, they found time to plan for the peacetime use of atomic energy that they foresaw so clearly.

Many industrialists looked upon radioactive isotopes as little more than a promising toy for esoteric research. Most experts on electrical power considered atomic electricity a dream that might come true in 50, 75, perhaps 100 years.

Yet, only five years later, radio isotopes were hailed as the most important instrument since the invention of a microscope.

Within ten years, electricity from atomic power plants was operating lights and toasters and radios in the homes of ordinary people.

The optimists were right, but we are just entering the Atomic Age. Noted atomic scientists say that only the surface of atomic energy has been scratched What changes the atom will bring in the future are unknown, only time can tell.

I'd like to thank Linda McDonald for allowing me to share her speech with you here today.

The headlines are full of the power needs across the country to power data centers in support of artificial intelligence. There are daily stories of industries returning manufacturing back to the United States, plants that need electricity to operate.

We have gone from shrinking demand for electricity to growth not seen in generations.

Utilities are deploying new generating assets to meet this rapidly rising demand. And when it comes down to it, there are only two forms of dispatchable base load generation that can be deployed, either a natural gas power plant or a nuclear plant.

Utilities strive for a diverse portfolio of generation to best weather unforeseen future events, like the war in Ukraine. Nuclear is one of the most resilient, reliable, and dependable sources of electrical generation.

TVA is leading the country in advancing the next generation of nuclear reactors and our work on the BWRX-300 at the Clinch River site is helping us lead that charge.

I'd like to thank the NRC, when you look at history, again, this is one of the first Part 50 applications in over 40 years. And many of the regulations have been modified over the years to really align to Part 52.

And that was one of our concerns three years ago when we started our construction permit application was, how were we going to navigate a Part

50 pathway that hasn't been used in 40 years with regulations that have changes.

So, I really want to thank the NRC. They've really worked with us. It's evidenced in the questions they ask, the engagements that we've had, the audit that we've had, that has really prepared us to submit what we hope is a high-quality construction permit application.

So, I first read this speech about ten years ago and it strikes me every time I read it how much progress we have made and the vast potential still ahead of us.

When we're in the middle of fray, we often don't realize that history is being made. But I think if you look around this room and at the innovation happening with nuclear energy throughout the country, I believe that we are making history that future generations will learn about in school.

Thank you for inviting us here today and I will turn it over to Ray Schiele.

MR. SCHIELE: Thanks, Scott.

Good morning, I'm Ray Schiele, Senior Licensing Manager for the TVA New Nuclear Program. It is a pleasure to present to ACRS again. Last time was for the full Committee the TVA CRN early site permit licensing effort.

Presenting to the ACRS today is not just a new chapter for TVA, but a continuation of the effort back in December of 2019 when the early site permit was issued.

On the slide we have on the screen right now is just a pictorial of the location of the Clinch River site as it relates to the Oak Ridge Reservation and the Clinch River arm of the Watts Bar Reservoir. It's 935 acres.

Next slide, please, Allen?

So, this slide is an overview of the journey to submit the CPA from the time our early site permit was issued.

The first thing we pursued was a technology evaluation. The early site permit was technology neutral. A plant parameter envelope was used to compare four designs of various levels of technical maturity.

The technology evaluation considered many feature including supply chain constraints, advanced manufacturing, seismic issues, modular construction, advanced construction techniques.

From that, while we were doing technology evaluation, we realized, looking forward towards the development of a construction permit, that Part 50 had not been done for some time.

So, we pursued developing annotated outlines comparing the content of Reg Guide 1.70, which is the standard format and content for nuclear power plants to NUREG-0800 to identify where gaps or inconsistences existed.

When the drafts of the annotated outlines were complete, we provided them to the NRC staff in the electronic reading room for review.

Staff comments were incorporated and the annotated outlines to further de-risk the scope of a construction permit application development.

As part of the technology evaluation, the BWRX-300 design was selected. And in June of 2022, TVA and GVH agreed upon a contract and path forward to develop a construction permit application for CRN-1.

The annotated outlines informed the development of the regulatory framework documents which further de-risked the development of a construction permit application.

Question?

MEMBER MARTIN: This is Bob Martin.

You mentioned for the 2019 ESP and the plant parameters envelope, was the BWRX-300 among those four plants?

MR. SCHIELE: No, it was not.

MEMBER MARTIN: Okay, so you've obviously done some looking to see if that envelope was broad enough?

MR. SCHIELE: So, that's correct. So, the four technologies evaluated were for PWRs of various labels and maturity. And in the early site permit, there are some tables towards the end that lists the parameters that enveloped the permit.

So, when we were doing our technology evaluation, yes, one of the things we looked at right away is, what technology at the level of maturity that it was at right now would fit into that -- those boundaries in the envelope there?

MEMBER MARTIN: And I mean, how much did level maturity matter? And I understand we're all some -- been around the block and we kind of know and followed TVA's progress on this stuff.

MR. SCHIELE: I think it was significant. One of the biggest things was fuel design. The fuel design and the supply chain assessment table was one of the biggest factors in selecting the technology at the time.

MEMBER MARTIN: That makes a lot of sense, supply chain's everything, isn't it?

MR. SCHIELE: Yes.

MEMBER MARTIN: Can you build it and can you operate it?

MR. SCHIELE: Exactly.

MEMBER MARTIN: All right, thanks.

MR. SCHIELE: Sure. So, CNR site -- early site permit was incorporated by reference where appropriate and the content was developed, and site specific content was developed like TVA estimates.

A freeze date of April of 2024 was chosen to control the scope of the application that we intended to submit to the NRC.

In parallel with PSAR development, TVA additionally developed a topical report for NQA-A for the site new nuclear. We developed and submitted an exempt request for a Part 2.101(a)(5) which further enabled us to leverage submitting for a Part 50 either the environmental report or the PSAR first gave us that option.

We also submitted an exemption request for early site excavation. Currently, as you know, the definition for excavation does not allow anything to be permanently left in the excavation because that would fall into the definition of construction.

However, for this deeply embedded containment and reactor building, we're talking about an excess of 120 feet in excavating and in stages, rock bolts, shotcrete wire would be left on the walls in stages to stabilize the wall with no really support for any safety features other than safety for the workers. And if those things were being removed, you'd further destabilize the walls.

So, we submitted an exemption request to allow those features to stabilize the walls to be left in and not be considered part of the construction.

CHAIR HARRINGTON: So, this is Craig.

I take it from that, then, you have already started digging the hole, basically?

MR. SCHIELE: No, we have not started digging yet.

CHAIR HARRINGTON: Okay.

MR. SCHIELE: This is all in preparation to dig the hole.

CHAIR HARRINGTON: Okay. Thanks.

MEMBER HALNON: Greg Halnon. You mentioned a freeze date of April 2024, what is that mean? What made you freeze?

MR. SCHIELE: So, the design is continuing to mature. And it's pretty impossible to keep an application in step with design maturity.

So, we picked a date that -- to ensure that the application was current with all the design features as of that date with the anticipation that, either part of the review process or afterwards where certain critical functions had matured, we would update the application with those things.

Now, there's been a couple exceptions to that. One is the isolation condenser system. We actually updated that because there was a design change on that. But I'd say 95-plus percent of the application was frozen to the design as presented in April of 2024.

MEMBER HALNON: Okay, so, you internally made a judgment that that design at April 2024 was adequate to get through at least the CPA?

MR. SCHIELE: That's correct.

MEMBER HALNON: And then, how will you identify the changes going forward in the FSAR here?

MR. SCHIELE: So, that's a good question.

As part of the annotated outlines we developed and the regulatory framework documents, content was cataloged DSAR or FSAR.

So, for those things that are cataloged FSAR, as we go forward in the licensing process and getting ready to prepare the operating license application, we have to ensure all those items that are coded FSAR are available to complete the application.

MEMBER HALNON: I would encourage you to continue to keep the review in mind, though, to the deltas between. We had some applications come in that did a very nice job with showing the deltas, the Kairos Hermes 2, for example. That's one way of doing it.

And we've had others come in that was just a brand new thing and it takes a lot more effort to review that to try to do a slew of delta documents.

MR. SCHIELE: Yes.

MEMBER HALNON: So, as you go through it, it'd be very good to keep that delta document or somehow --

MR. SCHIELE: No, 100 percent agree. In fact, one of the things we did for the construction permit, which we'll talk about in a few slides, was in Chapter 2, where it talks about tomography and meteorology and so forth, a lot of that information was incorporated by reference from the early site permit.

And we provided the staff markups for

those chapters of the information, redline markups in the reading room, to show what was IBR and what was new to support the review.

MEMBER HALNON: Yes, just keep the review in mind --

MR. SCHIELE: Yes, absolutely.

MEMBER HALNON: -- that'll help smooth it through.

MR. SCHIELE: Yes.

MEMBER ROBERTS: This is Tom Roberts, a quick question.

You wrote to emphasize this is the first construction permit application pertaining to Part 50 in 40 years, but there are two others that are currently in house, one of which preceded you by a year and as well as there's Kairos, which I know is a test reactor, but it's probably, you know, has some relevance to.

How much of that did look at? How much coordination was there in looking at what the other applicants were doing in terms of what the best practices were for the CPA?

MR. SCHIELE: We have been following other applicant challenges, specifically in the topical reports that they were submitting to overcome those

Part 50 issues, many of them legacy items from Part 52. So, we have been following the other applications of the challenges they've had.

MR. HUNNEWELL: So, I'll just add on that we have a cooperation agreement with Kairos power and we're actually actively involved in preparation of their application.

MEMBER ROBERTS: I guess the question for the staff maybe when it's their turn, presumably, they're also profiting from the -- y'all's interactions in kind of their own best practices. So, I'll just follow up on that when the staff's up.

Thank you.

MR. SCHIELE: Also, during this time period, in addition to exemption requests and the topical report, we prepared an environmental report per 10 CFR 51.59.

And also, TVA is a government agency, we had to do our own NEPA. So, we were providing -- we prepared a subsequent environmental impact statement taking advantage of what was done in the early site permit and processed that application.

As we were providing the -- or going through development during '23, '24 time frame, we requested the staff perform a readiness assessment per

LIC 116. The assessment findings were dispositioned in the application prior to us submitting a construction permit application.

So, finally, we get to CPA submittal. The construction permit application was submitted in two parts, the environmental report portion was submitted in April of 2025 and the preliminary safety evaluation report in May of 2025.

The acceptance review was completed in June and July of 2025 and the NRC commenced the review of the environmental report and the PSAR with audits.

The environmental audit is ongoing, nearing completion right now.

And the safety site audit is just getting underway.

MEMBER HALNON: And what's the schedule say?

MR. SCHIELE: So --

MEMBER HALNON: For approval?

MR. SCHIELE: Approval of the construction --

MEMBER HALNON: Yes, from the NRC when you're done?

MR. SCHIELE: Right now, I believe it's scheduled for issuance of the construction permit is

December of '26.

MEMBER HALNON: Okay, and that's what you got in you acceptance letter was 12/26?

MR. SCHIELE: Yes. Next slide, please. So, this is just a quick summary of all the regulations and Reg Guides that informed the various enclosures for our application. The application was over 4,000 pages.

Enclosure 1 is informed by 50.33.

Enclosure 2, public and private, Reg Guide 1.70, as we talked about, 0800, and 50.34(a) for our PSAR. And the environmental report was 51.50, NUREG 1555, standard review plan, and Reg Guide 4.2.

Next slide, please. So, this is a little more detailed look at the site. And the red on the left is the area of proposed disturbance.

And on the right is a high-level, but plant layout where we expect certain features to be deployed.

Chapter 1 summarizes the principle aspects of the design, conformance with the regulatory requirements, and material that is incorporated by reference in the PSAR.

Chapter -- Section 1.5 also provides, as allowed by Reg Guide 1.70, requirements for further technical information. Reg Guide 1.70 states that the PSAR should identify and describe and discuss those features or components for which further technical information is required in support of the issuance of the construction permit, but which has not been supplied in the PSAR.

Currently, we have two items in Section -- Chapter 1.5. The first are Appendices 3(b) through 3(h). These are the summary of the preliminary analyses that demonstrate the design of a seismic Cat I structure. Tentatively, we are going to provide those in a supplement no later than September of '25.

The second item is an evaluation assuming fission product release based on a hypothetical event. This is your containment performance analysis. It's being developed right now and we believe it will be submitted within six months after we submitted the CPA, so sometime this fall.

MEMBER HALNON: Ray, is the seasonal -- this is Greg -- the seasonal control, the river level going to be accurate the way it's done or do you have to modify that with the Authority?

MR. SCHIELE: So, that's a good question. The Clinch River arm of the Watts Bar Reservoir, it is a reservoir. It's not a free-flowing river. And the bypass flow from Melton Hill Dam, big topic of discussion during the that permit for what was proposed to be initially pursued there, there may have been a need for a bypass.

At this time, with just deploying a single unit, we don't believe a bypass is going to be required. But the river is not a free-flowing river. It's part of the bigger Watts Bar Reservoir, which is a controlled level between the two dams, Watts Bar Dam and the dams upstream.

MEMBER HALNON: This goes back a long ways, I don't want to put you on the spot if you don't remember, but one of your boreholes had -- they found diesel fuel in it.

MR. SCHIELE: Yes.

MEMBER HALNON: Can you remind me what the root cause of that or is it --

MR. SCHIELE: So, we did subsequent evaluations on that with TDEC, the Tennessee Department of Environment Control and we did additional sampling and it was determined that that was naturally occurring product. And those wells were subsequently closed. And we've annotated those details.

MEMBER HALNON: Yes, it was a big head

scratcher back then --

MR. SCHIELE: It was.

MEMBER HALNON: -- when it happened. And it almost felt like somebody just poured a five-gallon can of diesel fuel in that.

MR. SCHIELE: No, it was determined it was naturally occurring.

MEMBER HALNON: Naturally occurring, yes.

MR. SCHIELE: Next slide, please. So, Chapter 2 is site characteristics provides an evaluation against the early site permit PPE and justification of exceedances.

Site characteristics parameters, those items that were incorporated by reference from that SSAR, which is the site safety analysis report in the early site permit, early site permit conditions and early site permit COL action items.

On the slide you have in front of you right now, those portions have some or a majority of the content carried forward from the early site permit.

There's a table in Chapter 1 that describes the cross reference of all the chapter sections that have been incorporated by reference in the PSAR.

Next slide, please. In addition to updating many of the site characteristics, once we transitioned from the early site permit which had a footprint for the deployment of two reactors, the footprint for the BWRX-300 was identified and, per regulations, we had to have some conforming or some additional cohorts done at centerline and the four corners.

And then, additional core bores where the cooling towers would be and where the switch out would be. So, as part of Chapter 2, we've performed additional core bores for the footprint for the deployment for CRN-1. And for those of you were part of the early site permit, you'll remember that there was a tremendous amount of core bores from the breeder reactor and then, from the mPower reactor, and then, for the early site permit.

But in the picture, you can see the gray hashed markings, those are some of the ones from the previous efforts. They just didn't fall exactly where we needed them for four corners and center.

So, Clinch River is highly, from a subsurface perspective, highly characteristic.

Next slide, Allen?

MEMBER MARTIN: And you probably cover it somewhere, you mentioned the plan right now is for a single unit. But when you went into this with ESP were you're thinking multi-unit or, I mean, is there a potential to expand at some point? Or is this --

MR. SCHIELE: Well, the early site permit demonstrated that the site characteristics could handle the deployment of two or more with some certain boundaries, 800 megawatts electric.

MEMBER MARTIN: So, 800 --

MR. SCHIELE: Yes, 800 megawatts electric plus all the other parameters that would confine you to operating two reactors, whether it's the MET data or a heat to the river, all these things informed the early site permit. And that was for a PPE approach to a permit.

MEMBER MARTIN: Yes, right.

MR. SCHIELE: Going forward, we decided to deploy one, and Scott, you want to talk more about it?

MR. NGUYEN: Excuse me, Ray, can you speak directly into the mic? Thank you.

MR. HUNNEWELL: Yes, so, let me just touch on that briefly, right? So, the first thing is that TVA has not decided to deploy a reactor at Clinch River. The environmental report and CPA for a single BWRX-300 at the site, we are performing continency planning for the potential deployment of additional reactors at the site of either BWRX-300 technology or potentially a different technology.

So, from a planning standpoint, we're targeting a single reactor, but we are thinking beyond that.

MEMBER MARTIN: Okay.

MR. SCHIELE: Chapter 3, design of structures, systems, and components. Chapter 3 describes the classification of SSCs as well as compliance with the general design criteria and appendices that provide safety class, design analysis of seismic structures, and a computer program for design analysis of SSCs, and aircraft impact assessment.

On the slide in front of you is an illustration of how the reactor building containment would be constructed using DPSC, diaphragm plate steel composite method.

This chapter is also additionally informed by the safety strategy LTR which is in flight right now and being reviewed by the NRC, and also, the DPSC LTR, both of which I believe GVH will discuss later in their presentation.

MEMBER HALNON: Ray, this is Greg.

On the picture on the left, where's the current ground level on that?

So, you go, one two, three, four --

MR. SCHIELE: So, you see where it's the reactor building with wall the arrow right there, it's just above that, you'll see on the right hand of the reactor building, there's like -- do you see where the fuel -- spent fuel pool is?

MEMBER HALNON: Yes.

MR. SCHIELE: Just above that is ground zero.

MEMBER HALNON: Okay, so, then that area is completely under --

MR. SCHIELE: And I believe in the next presentation, David's got an elevation presentation.

MEMBER HALNON: Okay, good, thanks.

MR. SCHIELE: Okay.

MEMBER ROBERTS: I want to follow up, this is Tom, with a follow up to my question earlier about the first CP application for the area. And one thing I had mentioned I'd ask the staff a little later, but the staff's not here, so I'm not going to ask them, I'll try to follow up later with them.

What occurred to me is the other CPAs we have in are based on the TICAP, ARCAP advanced reactor concept application and there's some streamlining, significant streamlining that went into that.

And I was wondering if you'd looked at that when you considered using that structure instead of the existing in Reg Guide 1.70, you know, meaning light water reactor chapter, for instance?

It doesn't necessarily focuses is well on safety and your safety strategy and topical report?

MR. SCHIELE: No, we did look at that, but that was primarily designed for non-light water reactors and this is a light water reactor application.

So, a lot of the things in the Part 53 and LMP and TICAP and ARCAP really didn't apply. But there were some good insights on how we managed information demonstrating complaint and conformance for Part 50 in there for an advanced reactor.

And this isn't an advanced reactor, it's passive having been informed by safety strategy, yes. But it follows like the passive plant rules. Which, if you look at Part 53 and LMP and TICAP and ARCAP, it's the same information. It's catalogued a little differently but it's very close, very close.

MEMBER ROBERTS: Yes, this is probably something to look at in a couple years after testing these three CP applications to see if there's some common lessons in that. It maybe TICAP, ARCAP didn't exactly hit the mark, either.

But it just occurs to me that it has the potential to be more closely focused on safety and not have a lot extraneous material in there that's sometimes hard to integrate into the overall safety case. So, you know, more to come on that, but thanks.

MR. SCHIELE: Yes, thank you.

MR. NGUYEN: For clarification, Member Roberts, the NRC staff is not scheduled to present today.

MR. SCHIELE: Let's go on to the next slide, please. So, the BWRX-300 is a natural circulation on boiling water reactor. Chapter 4 describes the design of the fueling reactor, reactor core, including fuel rods, fuel assemblies, reactivity control system, nuclear design, and thermal hydraulic parameters.

There is an LTR being reviewed right now by the NRC for stability control that will inform this natural circulation reactor.

And I believe GVH will talk about -- more about that LTR in the next presentation.

Chapter 5 describes the reactor cooling

system and connected systems that form the reactor coolant pressure boundary.

The reactor coolant system is comprised of Safety Class 1 called SC-1 as far as safety strategy terminology goes, a portion of the nuclear boiler system and condensate feed.

The RCS extends to and includes the outermost containment isolation valves and the main seed and feed piping.

So, on this illustration, and I know it's just a one line diagram, is something that's kind of unique and interesting as compared to other designs.

The reactor isolation valves, and I believe David's going to discuss this more in the next presentation, but the reactor isolation valves for these major systems, feed steam and so forth, are also your in-core isolation valves on the vessel.

So, that's very interesting and that informs your reactor coolant pressure boundary with your isolation valves on the outside of containment.

Next slide, please. Chapter 6, engineering safety features, for the BWRX-300, passive systems that are not dependent on external source heat power or operator action and fulfill the fundamental safety functions for at least 72 hours after a design basis accident, there's three primary fundamental safety functions, control reactivity, removal of heat from the fuel, and confinement of radioactive material.

For the BWRX-300, it only credits three systems, containment, the passive containment cooling system, and the isolation condenser system.

So, on this illustration on the left, you'll see the three trains of the isolation condenser system and how they tie into the vessel.

And on the right, you'll see an illustration of the trains of the passive containment cooling system and how they tie in.

Next slide, please.

MEMBER HALNON: I'm sorry, Ray, this is Greg.

Did you say that the condenser -- isolation condenser system was added post-April 2024?

MR. SCHIELE: No, it was already in the design -- could you go back a slide, please. It was already in the design -- sorry, go to the Chapter 6 slide, please.

It was already in the design. What the design had was three independent pools for the three trains.

You see this illustration here, there's a pool for Train A, and there's another pool for Train B and C. So, the major change for it to go from three pools to two pools, so what you have right now is the pool for Train A that's guaranteed to support 72 hours, and the pool with two trains, 7-day pool.

MEMBER HALNON: So, that is an example of a significant modification that you updated the PSAR?

MR. SCHIELE: Exactly.

MEMBER HALNON: Okay, thanks.

MR. SCHIELE: And I'm sure David will talk about it more, but this is your primary ECCS system for 50.46, very important.

Let's go to Chapter 7, please?

So, Chapter 7, it describes the instrumentation and control systems used for normal, abnormal, and accident conditions.

Specifically, Chapter 7 includes the integrated digital based I&C design, the architectural arrangement that supports a plant-level defense-in-depth framework.

This framework relates to the safety analysis framework and the inner loop bases for the defense lines. Classifications scheme is based on importance of the individual defense lines.

The BWRX-300 I&C architecture and associated system components are designed with international standards and proven engineering design practices, we present state of the art methods.

This illustration is just a block diagram of the architecture and the trains that it supports and how it goes from the defense lines balance deployment and the data highways.

Next slide, please. Chapter 8, electric power, Chapter 8 provides the description of the alternating current and direct current power systems and power requirements for normal, abnormal, and accident conditions.

The electrical distribution system architecture is a configuration of generators, buses, transformers, and load centers that supply power to all the design loads.

The BWRX-300 design minimizes the reliance on electrical power support Category I functions. The passive design of the plant is not dependent on any AC power sources, including diesel generators to mitigate design basis accidents.

Safety Class 1 power is supplied from battery backed DC power.

The uninterruptible power system has a

coping period of 72 hours for design basis accidents.

In the illustration on this slide right there, it shows you two trains for a to batteries battery buses and the flows to supply the uninterruptible power system. It is the single source of power for you credited SC-1 design.

MEMBER ROBERTS: Hey, Ray, can you talk briefly about the control of those retro isolation valves? It seems like it's hard to call them as passive because I understand they need to shut to contain your inventory for subsequent design?

MR. SCHIELE: Sure, so, the SECY-94-084, the definition, the differentiation between passive and active is what provides the mode of force for a function to happen.

So, in this case, these valves, the RIBs, they are powered by stored energy, stored energy signal via the DC buses here.

It's not that the valves don't move or not as far as passive or active, it's where they get the energy to perform their function. And so, by the definition of passive, they perform their function using stored energy.

MEMBER ROBERTS: Okay, thanks.

But they rely on the batteries for mode of power and for sensing when to shut?

MR. SCHIELE: And from the I&C system to recognize that a certain event has happened. And not all the valves go shut, some of them are already shut, some will have to go shut to perform their function.

MEMBER ROBERTS: Right, but they don't shut on loss of power? Do they shut on loss of power to the I&C system?

MR. SCHIELE: I do not believe so.

David, can you --

MR. HINDS: Hi, this is David Hinds from GVH.

So, the reactor isolation valves, we have the -- those that I'll call power generation related that are all configured to fail in a closed position. They have stored energy such that if they lose power and/or control to that reactor isolation valve, the stored energy would cause the valves to go closed.

I will note that that power to them is battery backed, so that means loss of all power.

There is a set of valves for the isolation condenser system which are configured to fail as is.

I'll touch on that some more when I go through slide presentations, if that's okay.

MEMBER ROBERTS: Okay, thank you.

CHAIR HARRINGTON: And so, Ray, this is Craig.

I assume from the comments that there are no diesel generators at all?

MR. SCHIELE: No, there are diesel generators and the diesel generators are credited SC-3 power for those features after 72 hours like to replenish other systems. But they are credited after 72 hours.

CHAIR HARRINGTON: Thank you.

MR. SCHIELE: So, Chapter 9, please?

Chapter 9 describes systems used to support fuel storage and letting normal cooling water process auxiliaries, heat sink values and cooling fire power protection, power auxiliaries, communication light.

Appendix 9A presents the fire hazards analysis and the methodology for the fire safe shutdown analysis, both of which will be finalized and provided prior to the FSCRB issue.

On the slide right there is an illustration of the pool that has the fuel racks and the control blade racks in it. That's on the left.

And on the right, you'll see how it's part of the bigger scheme of pools at that level which provides your ultimate heat sink so you can see the Alpha pools for the condenser system and the Bravo, Charlie pools for the condenser system and the fuel storage racks.

MEMBER HALNON: Ray, this is Greg.

We haven't seen the word multiple in front of ultimate before. When -- does that mean that there's multiple or different heat sinks for different portions of the plant or is it a redundancy or how does that --

MR. SCHIELE: Sure.

So, for the isolation condenser system, the ultimate heat sink is the pools to atmosphere.

For the containment, the primary containment cooling system is another heat sink.

Both of those are for like modes, what you would call modes one through four or five.

But then, when you get into refueling, going back to that other illustration where you saw the very, very tall chimney, that whole column of water that would be filled up for the reactor cavity, that's the ultimate heat sinking mode which you would call mode six for refueling.

MEMBER HALNON: So, these are all safety class?

MR. SCHIELE: Dedicated bodies of water that will give you --

(Simultaneous speaking.)

MEMBER HALNON: -- the ultimate heat sink like a --

MR. SCHIELE: It sounds like the river, I don't know, Chesapeake Bay or something, you know,- or the ponds that look -- you know, some reactors had to build ponds next to them for their heat sink.

MEMBER HALNON: Is that unique to the country area or maybe, David, is that the way the plant was originally thought of relative here?

MR. SCHIELE: I think this is by design.

David, go ahead.

MR. HINDS: Hi, David Hinds, GVH.

So, the heat sink for safety is what Ray was describing. There is, of course, a heat sink for power generation. That's when we get to cooling tower and make up water.

Heat sink for safety -- I'll cover some more in the slides, but primarily focus on the isolation condenser system for the stored water within the safety Class 1 as to about safety Class 1 or safety related reactor building structure.

So, we store the water in the reactor

building. Similarly, we do the same thing for the fuel pool.

MEMBER HALNON: Okay, so, not unique to the site it’s just that --

MR. HINDS: No, it’s --

MEMBER HALNON: -- just this kind of set --

MR. HINDS: -- part --

MEMBER HALNON: -- so you can be nimble and place it anywhere in the country, then?

MR. HINDS: That’s correct, it is part of the standard.

MEMBER HALNON: Okay.

CHAIR HARRINGTON: This is Craig.

This drawing on the right, we have a similar version in GE slides, so I’ll ask it now, the inner and outer pools can you speak to how they relate to each other? Are they just communicated that it’s weir walls or what’s the details?

MR. SCHIELE: I don’t know, David, do you have that that shows the weir wall?

MR. HINDS: I’ll just describe it, we don’t picture that.

MR. SCHIELE: We may look it as if there’s a weir wall where, if one goes down, one will overflow into the other one and provide the cooling. Because effectively, you're steaming -- during an accident, you're steaming that water down. It goes out through the vents natural to the atmosphere.

So, the inner pools will feed into the outer pools as they steam down.

MR. HINDS: I do have a storage level figure when I get to that.

CHAIR HARRINGTON: Okay, good, so we'll have a better picture and talk about it some more.

Okay, thank you.

MR. SCHIELE: Slide Chapter 10, please?

So, the Chapter 10 describes the systems used for steam and power conversion. And this illustration is just a typical 1-HP wheel, 2-LP wheels, feedwater heaters, MSRs.

So, this chapter talks about turbine generator, main steam, and the associated system support for power conversion.

Next slide, please.

CHAIR HARRINGTON: This is Craig, again. On the power conversion, have you selected a vendor yet or is that still in progress?

MR. SCHIELE: It is not selected, that's why it's a very generic one. Chapter 11, radioactive waste management, Chapter 11 describes the capabilities of the plant to control, collect, handle, process, store, and dispose of liquid gaseous and solid waste.

There's two source terms that are discussed in Chapter 11. One is the realistic model which is based on nuclear concentrations that you would find typically in BWRs.

This model is referred to as the normal operation source term.

The other source term is the conservative design basis model. And it's based on GHGO clad defect and it's referred to in the chapter discussions as the design basis coolant source term.

Chapter 12, radiation protection covers the policy, design, and operational considerations for ALARA. Pretty straightforward.

Chapter 13, Chapter 13 discusses organizational structure, programs, procedures, staff qualifications.

For emergency preparedness, the early site permit had the major features of both a site boundary and a two-mile EPZ.

For a PSAR Appendix E of 10 CFR Part 50 requires a limited amount of information which is about the same amount of information we had in the early site permit.

So, we updated that information to conform to Appendix E and that's what you'll see in the application.

At this time, TVA has not decided to continue down the 15.47 pathway that the exemptions were provided for or to go down the 51.60 which is performance based. That'll be decided in part in the operating licensing application.

Chapter 13 also includes physical security and fitness for duty. As far as physical security, the security by design Reg Guide 5.90 I don't believe is quite approved yet. It's been through all the reviews. Again, TVA will evaluate security by design when it comes time to put together our operating license application.

MEMBER HALNON: So, on your operational, you see it being pretty traditional from the standpoint of the operations maintenance or is it -- or are you going to try something a bit more codec and to reduce staff to like on the BWR now?

MR. SCHIELE: The answer is yes, we're looking at all kind of aspects to lower the ultimate O&M costs post-staffing maintenance and so forth.

MEMBER HALNON: Okay, so you sort of started with the traditional and trying to figure down or are you coming from the bottom up?

MR. SCHIELE: You want to talk about this?

MR. HUNNEWELL: Absolutely.

So, we've got a targeted head count that supports our O&M costs going to LCOE. And that does assume certain things are automated.

For example, for work room management system, highly automated compared to today where you've got work room managers and schedulers who manually do everything.

So, there is -- and then, I've actually got somebody that is just over O&M reviewing the design with GVH to constantly give them feedback in areas where the design could be improved to help on the O&M side.

MEMBER HALNON: So, you're taking after the building in what you would love to have in your other plants now?

MR. HUNNEWELL: Correct.

MEMBER HALNON: Okay.

MR. SCHIELE: Chapter 14, initial test program. The initial test program is composed of phases characterized as construction, pre-op, and startup.

The construction tests serve as prerequisites to pre-operational tests which work its way up to your startup testing requirements.

Next slide, please. Chapter 15, safety analysis. The safety analysis provides information on the hazard analysis, deterministic safety, and probabilistic safety assessment.

The safety analysis scope includes normal operation, anticipated operational occurrences, AOOs, design basis accidents, design extension conditions, which you'll see that acronym, DEC, in the application, which is beyond design basis accidents.

Chapter 15 also includes two appendices, 15A which discusses the practically eliminated provisions, and also 15B which discusses complimentary design features for mitigating design extension conditions.

MEMBER MARTIN: Ray, I can probably guess the answer to this, but do you expect much of a departure on the NUREG-0800 Chapter 15? You know, obviously, the natural circulation, you know, probably brings in a unique stability event or something. But for the most part, it will kind of look familiar?

MR. SCHIELE: So, the licensing topical

report that GVH is processing right now, I believe will overcome most of those hurdles because when those safety evaluations are issued and we disposition the limitations conditions for lifesaving strategy, that should help manage most of those differences, which SSCs are credited for passive slant.

MEMBER MARTIN: I mean, go back 15 years ago when, of course, NuScale and mPower would doing a thing and they had a lot of design specific standards. I don't know, we don't have the staff here, you know, and although they're kind of here, but has there been talk of DSRS for the BWRX?

MR. SCHIELE: Not for Chapter 15. I mean, we used for Chapter 7 the design reviews DRG for Chapter 7 for instrumentation. So, that was used to perform the content of Chapter 7.

But I haven't heard any conversations about getting over Chapter 15.

MEMBER ROBERTS: So, the PRA, there's a lot of discussion certainly in the LMP world for the degree of quality that has to go into a CP of PRN. We had a meeting a couple months ago on how that applies to 10 CFR 50.

How much did you use PRA in the CPA and how did you resolve the, you know, the deficit quality questions?

MR. SCHIELE: So, as you know, a Part 50 application does not include Chapter 19 PRA.

However, we do have a Section 15.6 that has PRA information in it. And TVA participated with the staff in the initial discussion on the white paper on what is the scope of PRA for a construction permit application?

And as you know, there's an ISG that's been through you all getting ready to get approved that frames the scope for a construction permit or PRA.

Also TVA hosted in the electronic reading room a GEH design information on their PRA for the staff to review.

So, we've been very involved in the information sharing on PRA between the design that's already in the NRC. But the design right now, it's not mature enough to have even a preliminary PRA put in our application.

MEMBER ROBERTS: So, the motivation for that ISG was if folks were going to use PRA information in their application, but you're saying you don't really use PRA information at this point?

MR. SCHIELE: No, there are certain design features that are risk informed, yes. But there is not enough information to put together the initial PRA for a construction permit on a design that's still maturing. It would be different if we had like a design cert, but we don't.

MR. HUNNEWELL: Yes. So our approach to PRA at this phase is much more qualitative than quantitative. The quantitative, once the design matures to that point, is developed.

MEMBER ROBERTS: Okay. Thank you.

MEMBER DIMITRIJEVIC: This is Vesna. So did you use any quantitative information for your identification, categorization and grouping of the initial events and accident scenarios?

MR. SCHIELE: I'm going to have to defer that.

MS. BANKS: Good morning, everybody. This is Kelli Banks. I'm with GE Vernova licensing. So we do use also PRA techniques to prevent categorization. It's not that we're using the PSA model itself. So, for example, NRC has INL database where it has failure data, initiating event frequencies; so we use that type of information that is also an input to our PRA to determine the frequency of a given event. And we do categorize events according to frequency with some exceptions like LOCAs. For example, even if they are in a beyond design basis event frequency category, we still categorize those breaks as on-basis accidents.

So we use, I would say, similar information, but it's not that the PRA itself is being used. It's we use PRA engineers who are used to doing event sequence frequency determinations, and we use similar inputs.

MEMBER DIMITRIJEVIC: Thank you. But how about safety objectives and acceptance criteria?

MS. BANKS: So we do have preliminary appraisal to 15.6 for CDF and LERF. So based on, you know, the preliminary PSA that we have done up to the point, you know, that the design, when the PSAR was submitted, we do submit and show that, you know, the safety goals are also on track to being met. And, of course, once the design is finalized, the process for establishing a technically adequate PRA is finished, then those, you know, results would be also updated in the PSAR.

MEMBER DIMITRIJEVIC: All right. Thank you.

MR. SCHIELE: Thank you, Kelli. We'll go on to Chapter 16. Chapter 16, technical specifications. Chapter 16 provides the methodology for developing technical specifications and the associated bases that ensure compliance with the safety analysis inputs, assumptions, and results. It identifies preliminary variables, conditions, and items as a result of the descriptions of safety analyses contained elsewhere in the PSAR.

The selection methodology informs a preliminary table of contents, including the reason for inclusion of that selected content. The improved standard tech spec ISTS NUREG-1433 for BWR/4 plants and ISTS 1434 for BWR/6 plants were used as a template for the BWRX-300 tech specs and bases. A complete set of tech specs and bases will be provided as part of the operating license.

Quality assurance, Chapter 17. Chapter 17 describes the QA program used during design and construction of Clinch River 1 to ensure conformance with regulatory requirements and the design bases specified in the CPA. And, as I said earlier, we have a approved topical report that governs design, construction, and operation that is included in 17.5 of Chapter 17.

And the last thing is Enclosure 4. Enclosure 4 contains exemptions and variances that were included as part of the application. Currently, it has one exemption associated with reactor vessel material surveillance program requirements.

Variances. Right now, there are seven variances identified. These variances are based on -- there were over 40 COL action items in the early site permit. These seven variances are those items that the design put us outside the boundary for those items in the early site permit. An example is site grade level. The finished elevation in the site grade is 814.5, but in the finished elevation that was assumed in the ESP was 821. So we had to put a variance in for the difference.

So all of these are, for whatever reason, the variance, the difference, and the justification where we were outside the boundary of what was in the early site permit.

MEMBER HALNON: So some of those variances are actually conservative then to --

MR. SCHIELE: Yes.

MEMBER HALNON: -- for all the dirt you got to bring in.

MR. SCHIELE: Yes.

MEMBER MARTIN: I just wanted a clarification. I had asked a couple questions about the single unit and the PPE. Where you have the variance of the single unit thermal megawatts, talk a little bit about what that is.

MR. SCHIELE: So it was 300 megawatts electric. And this unit is -- David, you'll have to help me. I think it's what, 347?

MR. HINDS: I'd say nominally 300 electric.

MR. SCHIELE: That's nominal; but, actual, it's a little more than that?

MR. HINDS: So we had to justify why that's okay.

MEMBER MARTIN: Okay.

MR. SCHIELE: But the early site permit allowed for 800 for two or more, so we're still inside the permit, but we're higher than the single unit.

MEMBER MARTIN: Okay. That's interesting. So maybe you were anticipating another four PWRs were all smaller, but you went with probably the maximum power of the four.

MR. SCHIELE: Yes. And another way to look at it is that the early site permit justified a certain set of boundaries for the deployment of two or more. If TVA just chooses to deploy something else there, we would have to justify that application of that design separately. We couldn't use the early site permit.

MEMBER MARTIN: I took your 800 number and divided it by two. I'm going, oh, okay.

MEMBER HALNON: Ray, you know, you always think about, when you start building SMRs, you're thinking the end of the kind type thing. How close is this, what you submitted, do you think is going to be the next unit? I know Darlington's working on them, and you're probably informed by that. Did you take anything you see going into this PSAR something that may not carry through the rest of the nth of a kind?

MR. SCHIELE: Don't know that I can answer that right now. Darlington is first. It's the lead plant.

MEMBER HALNON: Yes. By all years, may be an nth of a kind --

MR. SCHIELE: We're watching Darlington closely.

MEMBER HALNON: -- even though it's a different unit.

MR. SCHIELE: But, as I said, in April of '24, we froze it. Lots of things have changed since then.

MEMBER HALNON: So we should expect a lot of red lines in the FSAR.

MR. SCHIELE: The FSAR will look definitely different than the PSAR. It's too early to say how it's going to change, but it will reflect the latest and greatest design maturity that we choose at the time.

CHAIR HARRINGTON: Can you say just a couple of words about the vessel surveillance program?

MR. SCHIELE: Yes. What we took the change on that was the ASTM year dates. So that was it.

CHAIR HARRINGTON: It's an administrative decision. I can't remember what all -- I've been reading, but I haven't got it.

MEMBER HALNON: Any other questions on the content of the application? Members, anyone online? Thank you.

MR. HINDS: Hi, I'm David Hinds. For GE Vernova Hitachi. Thank you for time today. So I'm here to give a brief overview of the design of the BWRX-300.

The BWRX-300, the acronym there, is, of course, the BWR. It stands for boiling water reactor. That's been our legacy at GE Vernova Hitachi. Sorry, new company name. But, anyway, so we have quite a legacy of design and fabrication, construction of boiling water reactors. So the X is Roman numeral 10. So this is 10th generation of boiling water reactor design in our evolutionary design. The 300 is just to represent the nominal electrical output. Of course, electrical output varies, primarily driven by cooling water of the specific site.

Go to the next slide, please.

MEMBER HALNON: Thank you. All right. Go ahead.

MR. HINDS: Okay. Thank you. So continuing on, just an outline to indicate some of the information covered. This is only a partial overview of the design with a focus on these areas. As always, questions are fine in any area, but these are areas that we've highlighted in the presentation and it also introduces some of the acronyms that may have filtered through the presentation. So talking about the reactor pressure vessel, or RPV. The reactor isolation valves, we've used an acronym RIV for reactor isolation valves. Isolation condenser system, or ICS, and the passive containment cooling system, or PCCS.

Go to the next slide, please. Thank you. As I indicated, the evolutionary design at GVH, we based many of our design decisions, design principles, analytical methods, codes, informed by prior generations. So I thought it would be appropriate here just to show a quick visual of some of that evolutionary design, and we noted and segregated the forced circulation design, which is in the green arrow here, and the natural circulation design in the blue or purple-looking arrow.

So the BWRX-300 is natural circulation. It is, though, informed by both forced circulation, and, when I say forced circulation, I'm talking reactor recirculation flow; and it's also informed by the natural circulation design predecessors. So some of the presentation will focus on those plants in the blue or purple.

Note Dodewaard is of particular interest in that it informed much of the design and operation of the BWRX-300. We did go through design certification of the ESBWR, which is also natural circulation, and we started a certification of the SBWR. So there's a significant amount of information related to the design and analysis and the regulatory treatment of the ESBWR from our design certification. So that highly informs our development of the BWRX-300. The sizes are different, but technology and principles are very similar.

MEMBER MARTIN: Could you -- this is Bob Martin -- speak to Dodewaard? Now, I just did a quick Google search, and I did know a little about it, but it looked like it was more of a demonstration plant, maybe an opportunity to do some interesting things with it and then collect data. Is there data that we might see at a later time that supports the eventual design certification?

MR. HINDS: Yes. So the bid for --

MEMBER MARTIN: There's juicier things, obviously, but it seemed like that was a huge opportunity, but it was a long time ago and things get lost.

MR. HINDS: Yes. Dodewaard actually operated for many years. I think it was approximately 25 years. It's in Europe and in Holland, and we gathered a significant amount of data, information, that informed our development of our analytical methods. Our TRACG computer code has a significant amount of benchmark data from the actual operation of Dodewaard. So it was very, very effective at informing and giving us confidence in our ability to design and operate a natural circulating reactor.

Dodewaard, as you can see by its coloring, it was a natural circulation. It's no longer in operation. However, it operated, as I said, for approximately 25 years. And, yes, the simple answer to your question is we did gather significant data, and that data is used to confirm our methods, analytical methods, design configurations are informed by the learnings at Dodewaard. The evolution such as startups, shutdowns, and power maneuvers are informed by Dodewaard. So, yes, it's very beneficial.

MEMBER MARTIN: Again, I'm kind of making assumptions about its use. Obviously, it was generating power. Now, would they have entertained tests that might have challenged the stability criteria design limits? And as far as what's, you know, most unique novel new here, you know, the circulation and stability questions are near the top of the list.

MR. HINDS: So, yes, we certainly gather data associated with stability performance at many different operating points within Dodewaard. Similarly, at even forced circulation plants, we've gathered stability data associated with it. Even though there are forced circulation and BWRX's natural circulation, the information is still relevant. So there are specific stability tests that were performed on a forced circulation power plant in Europe that highly informs our stability knowledge and methods.

So a combination of the forced circulation fleet, primarily in cases where there's some change in flow, such as stopping the reactor recirculation pumps on a forced circulation plant. But, yes, Dodewaard data highly informed our stability design or overall configuration of the reactor.

MEMBER MARTIN: Thank you.

CHAIR HARRINGTON: This is Craig. Not to turn this into a history lesson, but is there any particular reason or thought behind why there are two paths and why the Dodewaard plant was natural circulation?

MR. HINDS: Some of them get into business decisions. And I will say that the power density within the reactor is very somewhat driven by whether there's forced circulation or not. Generally, we have a higher power density for circulation reactors. So I'll say much of it is there's certainly technology feeders into that decision making, but there's also business aspects, too, of what's the desired output of a power plant, for instance; what's the limitation, whether it be business or technical or a combination thereof, on sizes of reactor pressure vessels. The configuration of the reactor pressure vessel is substantially or significantly impacted by the choice of natural circulation or forced circulation. So some are in the fabrication of components and many are business decisions but always informed by technology.

CHAIR HARRINGTON: Even utility or regulator influence, I suppose.

MR. HINDS: Yes. There's also operational differences and maneuvering differences between the different types of designs.

Can we go to the next slide, please. This slide has got many words, but it's meant to be, since this was a relatively short overview of something we could spend a long time on, we tried to pack quite a lot of information into this slide, so hopefully it's helpful to you. Some of these we've already touched on, or at least one or two of them. Okay. It's a nominal 300-megawatt electric gross output and, again, varies based upon cooling water. That's the power generation cooling water.

The reactor pressure vessel I mentioned, since it's natural circulation, the configuration of the reactor pressure vessel is substantially or significantly impacted by the choice of natural circulation. So the reactor pressure vessel height is selected accordingly. It's approximately a 27-meter tall vessel. It's approximately a 4-meter in diameter vessel. The diameter of the vessel is sized such that, based upon the core sizing plus the annular space for core flow, natural circulation, highly informed by fire-operating plants.

We have 240 fuel bundles. It was mentioned by Ray that part of the decision-making was the availability and maturity of the fuel for the reactor. We're using Global Nuclear Fuel, GNF-2 fuel, which is very highly experienced fuel. It's currently being manufactured and operated today, so it's not a new fuel development. It is fuel that's already proven, and so that was a very strategic choice.

The GNF-2 fuel, out of the various GNF fuel product lines, GNF-2 was specifically chosen, one, because it has a history and it's been proven, but also because it has favorable natural circulation behavior, low pressure drop, pressure drop that's acceptable for a natural circulation reactor. So it was a nice synergy, experienced fuel, currently manufactured, proven in the industry, and the pressure drop characteristics matched what's needed for our natural circulating reactor.

CHAIR HARRINGTON: This is Craig again. Are the fuel bundles standard length?

MR. HINDS: Yes. Very good question. If any of you have studied or been involved in our ESBWR, we did, in the ESBWR, just going back in history for a minute, we actually chose a special fuel design for ESBWR where we reduced the height or length of the fuel bundles for the purpose of pressure drop characteristics. What we found by selecting GNF-2 fuel and part of that strategic objective of not developing new fuel, we were able to use the standard fuel length. So simple answer is standard fuel length.

MR. HUNNEWELL: It's the same fuel that we use at our Browns Ferry reactor.

MR. HINDS: We have 57 control rods. I mentioned briefly that this actual core configuration. I'll show a core here in a little while, but it was also informed by an operating plant, KKM. I might mispronounce it; it's in Europe, but operating reactor, which is very, very similar core design. It was a forced circulation plant. However, the core design, highly informed by that. So a lot of history.

MEMBER HALNON: Are the control rods relatively standard? It's probably not a great word for it --

MR. HINDS: Yes.

MEMBER HALNON: But proven technology.

MR. HINDS: Yes.

MEMBER HALNON: You don't have to do cycle testing to see how many cycles or anything.

MR. HINDS: They are proven control rods. So we were very selective about the introduction of new features that haven't been used in the industry. Fuel and control rods are not new features. They are used in the industry.

So simple answer, yes, the control rods are quite similar to the exact same as our Marathon-Ultra type of control rods. So they are proven in the industry, so no new introduction of control rods. There is a slight adjustment between, depending upon which plant we're comparing to. We have a slide coming up, but this plant uses fine motion control rod drafts, so it has a special coupling. So the coupling is different between this plant design and a locking piston older plant design.

But outside of the coupling design, it's the same. We have history with the coupling as well.

Passive design. This is a passive safety power plant. So natural circulation and passive safety. And we've found that there is a very nice synergy between the passive safety and natural circulation.

As I display here with the height of the reactor pressure vessel, there is a need for, I'll just say a relatively tall reactor pressure vessel to support the natural circulation and to create the flow.

There's also a synergy with our passive safety with the coolant preservation approach, which I'll touch on through upcoming slides. So we have a lot of coolant relative to our past designs of forced circulation plants. A lot of coolant is already in the system, so that is a very nice synergy with passive design and the 72-hour requirement associated with passive power plants.

It was mentioned by Ray that we do not depend upon electrical power for safety. Actually, we do have diesel generators, but those diesel generators are not credited in our conservative safety analysis. We do have DC-backed buses. The highest safety class 1 DC buses are credited in our safety analysis. However, we configure it such that, even upon loss of DC, the plant is ensured to be safe. So we have this plant such that, in the end, if loss of all electricity, the plant will be in a safe configuration. We'll talk more about the safety features here in a minute.

So some other key features. I've already mentioned some numerous times here already natural circulation. I'll show a figure of the reactor pressure vessel, but a key feature to support natural circulation is the chimney. So I'll talk more about that.

The coolant preservation approach which I briefly touched on. So we have more coolant in this plant because it is natural circulation in this reactor. We also introduce a coolant what we call coolant preservation approach where we strategically place nozzles on the reactor pressure vessel relatively high, well above the top of active fuel, so, thereby, improving the ability to cool the fuel even in the event of a loss of coolant accident. So we remove by design threats to cooling the reactor by doing that.

And mentioned the reactor isolation valves. I'll show more in upcoming slides. The containment is a dry containment. It is nitrogen inerted and it does have passive cooling. Mentioned the acronym SCCV, Steel Plate Composite Containment Vessel. So the construction of this plant is a steel concrete steel, steel plate, diaphragm plate, or DPSC, Diaphragm Plate Steel Composite construction. That's both for the containment and for the reactor building.

MEMBER HALNON: Can you tell me what the approximate volume of the containment is?

MR. HINDS: It's a little less than 7,000 cubic meters.

MEMBER HALNON: That's a couple orders of magnitude less than the big ones that we're used to seeing. I think the standard PWR is around 2 million cubic feet.

MR. HINDS: I can't quote off memory the PWR standard.

MEMBER HALNON: Yeah, it's anomalous, at least the ones I've worked on.

MR. HINDS: Okay. Thank you. Overpressure protection is provided by the isolation condenser system along with the reactor scram function. We have submitted LTR on that which I'll bring up in upcoming slides. Emergency core cooling is performed by the isolation condenser system in conjunction with the reactor isolation valves. Some of the upcoming figures will help reinforce these last two bullets because they're quite significant bullets.

Can we go to the next slide please?

MEMBER ROBERTS: Yes, I guess we'll get

into some of this with the detailed picture. This big picture, it seems like the safety system is predicated on shutting RIVs quickly on a leak and not shutting isolation condenser valves on almost any scenario; is that right?

MR. HINDS: That's a good observation, and yes. So, in accident configurations, for example, if we were to postulate a loss of coolant accident, then the configuration would be similar to what you described or the same as what you described, meaning power generation-associated piping systems connected to the reactor vessel are automatically isolated. Isolation condenser system is placed in service to provide cooling and pressure control. So, yes, you're correct.

MEMBER ROBERTS: What kind of prototype response are you assuming for the isolation valves?

MR. HINDS: The isolation valves, we have an overall 15 seconds credited. That includes sense, you know, the signal to be generated, to be sensed, to go through the IEC platform, and then the closure function. Closure function is generally credited to be around five seconds once all the commands, once the sensing command is all complete.

MEMBER ROBERTS: Okay. I guess it goes

without saying you've tested that with the various DPs you will have during the event.

On the isolation condenser valves, are they serving a containment isolation function?

MR. HINDS: So the isolation condenser system valves, as with other reactor isolation valves, they serve a dual function of reactor isolation and containment isolation. I'll note that for the isolation condenser system, there's very, very little of that system that goes outside of containment. It goes straight up through the top of containment to the heat exchanger and straight back. So the only portion that's outside of containment is the heat exchanger immersed in a pool.

But the answer to your question is the reactor isolation valves serve a dual role of containment and reactor isolation.

MEMBER ROBERTS: Yes. I think with the isolation condenser valves, obviously, the lesson we learned from Fukushima is there can be conflicting design requirements on needing to keep them open for decay removal and for containment isolation, and that's part of the longer list of problems that caused that accident. If we could sort of talk about that later, about how you'd meet the dual requirement of shutting them for containment isolation and keep them open for decay removal?

MR. HINDS: I'll touch on it now, but I'll try to touch it on again. But very good observation. The isolation condenser system performs a high safety significant function. We protect that function. And keeping the system in service when it's needed is highly important.

So, yes, there is also containment and reactor isolation function. The high priority safety function of that system is to cool the fuel. So the prioritization is that the system will cool the fuel, and the design is accordingly. We do have leak detection and isolation functions if there were to be a loss of coolant via that system. But there was a high priority given to within the design of ensuring that system remains in service. That's a safety function.

MEMBER ROBERTS: Okay. Obviously, the devil is in the details on this one, but it just seems like it's a challenge going in to have safety functions that are basically conflicting and having to manage both of them.

MR. HINDS: I understand your question. It was a very good question.

This slide is a busy slide, but I'll call it a brief overview of our defense in depth and safety strategy approach. We have a safety strategy licensing topical report with the NRC, and this is a graphical or figure representation of much of the information within that. I'll give a brief overview, which should be coming through -- there will be further communication associated with the safety strategy since it is currently under review.

So we've taken a very, very rigorous approach to defense in depth on this plant design from the beginning. It's not an overlay or not an afterthought. It's embedded in the design, and this figure helps to represent.

So at the top we have physical barriers that we're protecting. That's the fuel cladding, your reactor coolant pressure boundary, and the containment. Then we have what we call defense lines labeled there in the center of the figure. The defense lines 2, 3, 4A, 4B, that's the way we stack up our defense in depth approach and we align within the design functions that are assigned to defense lines to ensure that those physical barriers are protected and maintained and that the fundamental safety functions of fuel cooling, confinement of radioactivity material, and radioactivity control are maintained.

So we rigorously go through postulations of events and ensure for each and every event that's postulated that not only do we have a defense line function, such as an actuation of a reactor scram for reactivity control, not only do we have function within a defense line to defend against that threat to safety, we also have a defense in depth measure such that we also, at the same time, are protecting from failures within our safety functions, such as a common cause failure. It's built in to our designs, so we can withstand a common cause failure of a control's platform, for instance. It's built in to this layered approach, such that we have two layers of defense to the event sequences that we've postulated that begin with a postulated initiating event. We characterize those events into categories of AOO, or anticipated operational occurrence, design basis accident, or design extension conditions. So we cover the whole spectrum of event sequences from the more frequent to the very infrequent. And we apply this approach to ensure safety is maintained if that event were to occur.

The event sequences are layered such that we postulate the initiator, then we take the same event and we'll postulate failure of a mitigating feature to prove effectiveness of the next line of defense within defense in depth. Defense line 1, shown in a couple of places on this figure, is a way that we capture robust design to start with, such that our goal is we don't even want the event to occur to start with. But if the event were to occur, these defense lines 2 through 4B are there to defend against the event and ensure safety is maintained.

A lot of information on this slide, but it is a very, very rigorous approach of defense in depth. I would say the functions that are assigned to those defense lines, we also apply design rules to them and classification rules. So the safety classification of these different defense lines varies.

The one in the center, defense line 3, is what's been safety category 1 or what's analogous to safety related. But we can't ignore the defense lines 2 and the 4A and 4B, which also have a safety classification using a classification approach that's aligned well with the International Atomic Energy Agency and other programs within the industry, such that we have a safety category 1, safety category 2, safety category 3, and a non-safety category within our design. And that flows all the way through the design, as well as the procurement of the associated components.

MEMBER HALNON: Quick question then. These are all just design features, not operational programs, correct? Or do you have operational programs embedded in the defense lines?

MR. HINDS: I'd say much of the operational programs would be, I'd characterize as embedded in the defense line 1 in that the operational programs, as well as the maintenance and other programs, help to ensure that the plant is in a proper state for operation and is a robust design and operation. And that underlies part of our thinking of minimize the event to start with.

Now, there are programs also that are graded on the associated quality programs and controls programs for the procurement of components within each of the physical defense lines, as well.

MEMBER HALNON: So it's sort of cross-cutting defense?

MR. HINDS: Yes.

MEMBER HALNON: Is the operator credited for any LDE incident?

MR. HINDS: So the operator is not credited for design basis accidents. Our design has been that if we see that -- if we postulate an event, event sequence, and, therefore, we design a defense line feature to mitigate that event, it is never the operator. It is a feature within the design. Of course, the operator is important to be there and is part of the overall insurance that the plant has operated within its expected operational bands. However, the defensive measures that I'm labeling here, defense lines 2 through 4B, those are design features that are highly automated or fail safe and fail in a safe state, not the operator.

MEMBER HALNON: Okay. Thanks.

MR. HINDS: Next slide, please.

CHAIR HARRINGTON: I think this is probably a convenient time to take a short break. We do have one question from Dennis Bley, I believe. We'll do that before we break. Okay. Dennis.

DR. BLEY: Thank you. You kind of already answered this question, but I like, I think I like the approach you're taking, and it struck me it's a very similar approach IAEA and others in Europe have worked on, and I think you said that.

That approach also looks at each level of either lowering the likelihood of getting into the state you're in or reducing somehow the consequences at that point. Is that right, that it's an evolution of what they did over there, or is this something you develop more on your own?

MR. HINDS: Yes. You're correct that it is heavily based upon the IAEA approach. However, there's very much work that we've done in order to develop our implementation of that because, in many cases, the guidance programs that we're both referring to from IAEA are a little more general than the specifics that we need to design and analyze the power plant.

So we developed a significant amount of detail to come up with the implementation program. And like this figure, for instance, we developed informed by IAEA, but we created this program and figure. But I'd say it's very, very well aligned with the IAEA.

DR. BLEY: Okay. Thanks. I've always liked that approach, and I'm glad you found a way to adapt it to what you're doing.

MEMBER HALNON: Dennis, you have a squeal on your line. If you're on the same computer, I suggest, during the break, you log off and log back in. Maybe they'll fix it.

DR. BLEY: Okay. Thanks. I have a

problem with this computer as the NRC one. If Quynh could send me another invitation so I could use my other computer, that would help.

CHAIR HARRINGTON: Okay. We'll let this this slide soak in for a few minutes. So we'll reconvene at 25 after the hour. Thank you.

(Whereupon, the above-entitled matter went off the record at 10:11 a.m. and resumed at 10:29 a.m.)

MEMBER HALNON: So, David, since we lost the record there for a little while, could you start back up on the first LTR?

MR. HINDS: Yes. So this is a brief summary of the licensing topical reports that have been submitted and reviewed and then, in further slides, we'll go through licensing topical reports that are currently under review. So this slide has licensing topical reports that have been submitted and already reviewed.

The first LTR was NEDC-33910, and that focused on reactor isolation, reactor pressure vessel isolation function and overpressure protection function. And in brief summary, this function was to isolate the reactor, including there would be a reactor scram that would occur, reactor scram, isolation, and isolation condenser systems come into service, overpressure protection would be performed by the function of the reactor scram plus isolation condenser system cooling. I'll note the isolation condenser system has a substantial capacity such that it can handle the cooling function of decay heat, including those of a pressurization transient. We'll go through that a little more on an upcoming slide of isolation condenser, but this was all introduced in the 33910.

The next LTR is 33911, or NEDC-33911, move from reactor boundary to the containment boundary. We introduce the type of containment, which is a dry nitrogen inerted containment with passive containment cooling, and also included in this LTR was the containment isolation function, which includes an integration between an outside reactor or containment isolation, coupled with the inside containment isolation being performed by the reactor isolation valves. This was all introduced in the 33911.

The next topic introduced by the LTR was NEDC-33912. Yes.

CHAIR HARRINGTON: This is Craig real quick. For the piping segment from the RIVs to containment moment restraint, do you have to treat that piping differently in any way?

MR. HINDS: Well, I'll say that piping is very specially designed to special stress rules. It's extremely robust as such that it would meet any rules associated with, for example, break exclusion zone, which is part of the French technical position with the U.S. NRC. It meets those rules, so maybe that's getting to the point.

CHAIR HARRINGTON: Yes, that's what I'd suspect. Okay.

MR. HINDS: The 33912 LTR was reactivity control. Much of the reason for this LTR and the focus is on the means of reactor shutdown. And this plant uses fine motion control rod drives, which have both a hydraulic insertion method and a motor insertion method. This design and the associated controls and platforms associated with it, giving us two means of shutdown via fast reactor scram and a fast motor run-in, along with the controls platform, since command and actuate type of platform, to give us protection mitigation for an assurance of a reactor shutdown when needed. And compliance was addressed associated with the ATWS rule for the U.S. NRC.

And I'll note that we did justify and credit those two means of control rod insertion. I'll note from a design perspective, I think most of you are probably very well familiar with the history of the hydraulic scram functions. For the motor run-in function for this plant, we did include some special design features such as the speed of the motor is such that it gives effective mitigation even in the case of loss of the hydraulics if we have a common cause failure. And we did include a UPS system for insurance that power would be available for that second means of shutdown. Topics of reactivity control were addressed in this LETTER.

Next slide. Okay. This is continuing on into already reviewed LTRs. So the first one on this page is 33922, which is containment evaluation methods. If you remember on the prior slide, there was one on containment functional performance. This continued all in the theme of and more detail associated with how the containment is designed and analyzed. So this licensing topical report introduced the analytical methods and qualification of those analytical methods for containment performance in the presence of a design basis accident. So this LTR built upon what was already introduced in 33911 and 33910 focused on the containment.

The next LTR was for civil structural

area, and it was 33914, advanced civil construction and design approach associated with the reactor building design. It's an embedded reactor building design with a Diaphragm Plate Steel Composite construction. So those aspects of embedding the reactor building in the ground and being constructed with the steel plate and concrete composite construction, the associated design requirements and analytical methods were introduced in this licensing topical report.

MEMBER MARTIN: I'll ask a question here on your containment evaluation methods. As you noted early in your presentation, you've been through this with the ESPWR, ESBWR. So really much departure from those methods that you introduced, I guess, over a decade ago.

MR. HINDS: We did change the methods. So there's an evolution. We changed the design, which prompted us to change the methods. So, in ESBWR, we actually use TRACG as the analytical tool for both reactor cooling system and containment. And, as I think you're probably aware, the ESBWR had a wet containment, pressure suppression, suppression pool type containment. BWRX-300 has a dry containment, no suppression pool.

That change in design type prompted us to revisit our analytical methods, so that's what we introduced in 33922. We transitioned from the TRACG tool for containment. TRACG is still used on BWRX-300 for the analytical tool for reactor coolant system. But for the containment performance, we are using GOTHIC as far as the primary tool. The mass and energy release comes from TRACG and is handed off to GOTHIC for the response that the containment performs.

MEMBER MARTIN: Pretty standard in that sense, yes.

MR. HINDS: It's recognized tools.

MEMBER MARTIN: Yes, exactly.

MR. HINDS: We just selected a different one than we had in our predecessor designs that we introduced at the LDR.

Next slide, please. Okay. Now we're into licensed topical reports that are still under review. The prior ones, just for a reminder, have already completed their review. So the first one of these is 33926. And since these are still under review, I would assume you'll hear more.

The one on the prior slide that I mentioned that was civil structural oriented was for the reactor building Diaphragm Plate Steel Composite construction. This one's focused on the same type of construction of the Diaphragm Plate Steel Composite, but on the containment is included in this LTR. And so it really builds upon what was already introduced in the prior page LTR, but it gets more into the containment, as well. You can see within here the bullets of what's introduced, the design approach and methodology, materials, fabrication, many topics covered, and the associated criteria for this seismic and ASME. From the containment perspective, seismic and ASME structure/component. Functionally, it's a component; but, physically, it's a structure.

So this is all still under review, so I didn't intend to go very deep into this since it'll be more introduced as it continues through the review. It looks like Ray has a comment maybe.

MR. SCHIELE: For Greg, to answer a question about where zero was? If you look on there, at that picture, zero's right at the bottom base of the fuel pool. If you look on the right-hand figure, see the red floor? That's a green.

MEMBER HALNON: The thicker one.

MR. HINDS: Yes, the thicker one. Yes. The top red floor. So, basically, the majority of the containment is embedded underground, as you can see by what we just discussed, and the majority of the rector pressure vessel is underground, and the pools are above ground.

I'll note that while we're on the figure also that with the pools above ground, but they're near grade; it's very easy to get water in them. They're for refill. These pools are non-pressurized. So just in your thinking about defense in depth and, you know, eventually a pool needs refilled, it's not very hard to get water in these pools. They're very close to grade elevation, and they're not pressurized.

But, anyway, I'll just highlight that, and also I'll highlight a couple of other things. Okay. So the green here is just colored like that to show the outline of containment, and the red is the reactor building structure around the containment. They're both cylindrical, cylindrical containment inside of a cylindrical reactor building, and we just spoke about the embedment.

The refueling area is that area up in the top, of course, and in those pools, many of the pools up top have, from the pool it would look like a roof, but from above it would look like a floor and it serves as refilled floor. So the isolation condenser system pools are underneath concrete so, basically, underneath the refueling floor. The fuel pool is an open pool. So the isolation condenser system pools are again covered by concrete, and the reactor cavity pool is an open pool which is right above the reactor pressure vessel. You can see here it's got blue there indicating there's water in it and it does have water in it. It's normally a flooded reactor cavity as opposed to a dry reactor cavity.

MEMBER MARTIN: So these pools, so, obviously, the boundaries are concrete, but they're steel lined? Is that the intent or what's the support for the pool?

MR. HINDS: Yes. The structural support is concrete. Concrete, yes. And then there's appropriate liners, where appropriate, for liners to ensure that there's no, you know, there's watertight boundaries. Exactly, yes.

CHAIR HARRINGTON: So more than just the steel composite plate, steel in their surface, there would be a pool liner in addition to that?

MR. HINDS: Where appropriate. So, for instance, in the fuel, yes.

CHAIR HARRINGTON: And with the concrete roof, floor, over those pools? I guess there's then a vent path forward.

MR. HINDS: Exactly, exactly, yes; you're correct. So I'll have another figure on the isolation condenser system. It may help highlight some of these features. I'm just trying to show the physical layout within the building, but your statement is correct. The isolation condenser system pool surface is vented to atmosphere and there is, as we both said now, a roof over and there's a vapor space above the water surface and that vapor space is vented.

CHAIR HARRINGTON: Dennis has a question, but, first, let me ask one other clarifying thing. On the earlier cross-section of this area of containment, there's one segment that was not marked as a pool. I guess that's access for shipping fuel in and out and things like that.

MR. HINDS: Oh, I understand your question now. Yes. There is an area that allows for access from grade level into the structure for, like, for instance, one of the figures that Ray had shown has showed a cask in there. So a spent fuel container cask can be moved and loaded within the pool, lifted, moved out via that access, for example. And there's also access such that equipment can be moved to and from the refueling floor.

So, yes, you're correct. There is an

access area in one of the quadrants, and there's also a personnel access area over in another quadrant. But we do take up a lot of real estate of the upper portion of the building near grade with water.

And back to the heat sink question you asked earlier, the safety heat sink you're looking at, the safety heat sink is all protected within that safety category 1 or safety related structure. The power generation heat sink is the one that's outside. So it makes for quite a robust protection of ensuring the cooling is maintained for safety. It's all inside the structure.

CHAIR HARRINGTON: Dennis.

DR. BLEY: Yes. Over on the right side, just below the 15-foot level, there's what looks like a penetration. Is that a way to bring outside water in an emergency, or what is that?

MR. HINDS: I was having a little trouble seeing the -- okay. I see where you're looking. There are some penetrations up above grade. For example, HVAC has to come in and out of the buildings. We do have penetrations for HVAC. We do have other service penetrations that aren't shown here for piping and cables. There is, since you're asking about penetrations, I'll also mention there is a steam tunnel area that's a quadrant where it's right up at the top of containment. On this representation, I believe it'd be over on the left side of the figure, just underneath the pool surfaces. There's a dedicated room there that's radiation controlled where the steam and feed water pipes, for example, exit containment, go through a steam tunnel, and go out to the turbine building.

But I believe the penetration you're pointing to, I believe, is HVAC. It's just representative penetration.

DR. BLEY: Okay. And they just don't show what's going on inside. Okay. Thanks.

MR. HINDS: Okay. Can we go to the next slide, please.

MEMBER MARTIN: Where is the load to support the reactor vessels? What level is it at?

MR. HINDS: The reactor vessel is supported by a pedestal structure. If you see the blue --

MEMBER MARTIN: Yeah.

MR. HINDS: -- RPV pedestal shown there. It's a cylinder. So, we have lots of cylinders here. So, we have a reactor building cylinder, containment cylinder inside of that, and inside of the containment cylinder is another cylinder, the reactor pressure vessel cylinder, and the reactor pressure vessel sits down inside of that cylinder.

It has a dual function. It provides shielding and it also provides support of the reactor pressure vessel. There's a skirt assembly you can see up about mid-height on the reactor pressure vessel, which is part of the reactor pressure vessel assembly, and that's where the connection is made between the reactor pressure vessel and the pedestal, and the pedestal is supported down on the base mat.

MEMBER MARTIN: So, like in your seismic analysis --

(Simultaneous speaking.)

MR. HINDS: Yes.

MEMBER MARTIN: A lot of attention there.

MR. HINDS: The pedestal is a very big focus area in the seismic analysis, yes. Good question.

CHAIR HARRINGTON: And this is Craig again. I guess the little gray circles there and down below are airlocks?

MR. HINDS: Yes, we do have accesses to the containment, and we have airlock access. We have equipment. We have upper and a lower access, and so, yes, you're pointing at the upper and the lower.

Upper is focused on doing maintenance on valves and other components of the upper portion of the containment, and the lower is focused on doing maintenance on under-vessel components normally sealed and closed as part of containment boundary during operation. Okay, next slide, please.

This is, I believe, the last one of the licensee topical reports that we have listed here for under review, and this -- no, I'm sorry, there's one more, but anyway, this is the safety strategy. The safety strategy is still under review.

I introduced that on the prior slide with that figure with the defense-in-depth. Those types of concepts are introduced in this safety strategy LTR, and then there's a regulatory evaluation associated with our design and analysis associated with that approach that's introduced in this licensing topical report.

As I mentioned before, we do have a graded safety class, Safety Class 1, 2, and 3, and on safety, and we made the connections between the design analysis and the regulatory evaluation associated with that defense-in-depth approach, and this includes evaluation against the GDCs, for example, so that's all embedded within this safety strategy LTR still here under review.

CHAIR HARRINGTON: And so, the implementation of this will be reflected in the FSAR or will there be another separate report that captures that?

MR. HINDS: In all aspects of our design information, and analysis information, and then the applications submitted by TVA define aspects of this safety strategy embedded within there, the terminology used and the classification of SSCs. It's already embedded in there.

CHAIR HARRINGTON: Okay.

MR. HINDS: This gives the wrapper, if you will, of introducing the whole concept, and the process by which it's treated, and the regulatory evaluation such that it's in a focused type of review as opposed to scattering the review only within the application.

CHAIR HARRINGTON: That's about where I am in the PSAR, so I haven't seen that, okay.

(Laughter.)

MR. HINDS: Okay, next slide, please. Okay, now I think we're to the last one. So, this is also under review, a stability analysis licensing topical report.

We introduced some concepts associated with stability or the regulatory treatment of stability within the reactivity control LTR that I had previously mentioned of 33912, but this one goes deeper and gets into the stability analysis of this plant. It's a natural circulating power plant, so many, including us, focus on ensuring thermal-hydraulic stability is maintained within the plant.

Just a couple of notes on stability, I mentioned previously in the other side that the core design heavily leverages the prior design and operation of the KKM core configuration, the same number of fuel bundles and core configuration.

It leverages the learnings from the prior natural circulators, and the analytical methods are built upon that. Also, I have test and development programs that further feed into that. So, the specific topic of stability and the associated analysis is introduced in this licensing topical report.

The core design that I've mentioned several times here, its behavior is such that it's very tightly coupled and we do not see threats to regional instability in these. The core is not big enough.

If we don't see threats to regional instability, I'll contrast it to the larger ESBWR, which has already been reviewed by the U.S. NRC. We did focus on regional stability on the ESBWR, but we found with the size and the type of core we selected for this plant, it's very closely coupled and regional stability is not an issue.

And we present those concepts and topics within a combination of the construction permit application, the 33912 which has already been removed, and this LTR here, but it behaves quite well with stability. I've got some more topics, or more tidbits on that that will come up in some future slides.

MEMBER HALNON: So, you're not going to be worried about project power until the last stage, and we think we're getting away from the mid-cycle readjustment of control rods.

MR. HINDS: We're not ruling out rod sequence exchange if that's what your last --

MEMBER HALNON: Yeah, yeah.

MR. HINDS: So, there will be some rod pattern changes within the cycles. So, we have been doing quite a lot of analysis with core design, including the entirety of, you know, going through a cycle, and so there would be some rod pattern changes within the cycle.

It's quite simple rod patterns on this plant though. Primarily, we use a control cell core, and primarily, it's four control rods doing --

(Simultaneous speaking.)

MR. HINDS: Well, that's what's doing the -- in control at any one moment.

MEMBER HALNON: Okay.

MR. HINDS: Now, at another point in the cycle, those four, we swap to a different four, but it's typically two to three groups of control rods. We have them in groups, two to three groups of control rods primarily with groups of four, and the center rod plays in some, that are actively in control. It's quite simple.

MEMBER HALNON: So, you say you're trying to design those out or at least minimize it?

MR. HINDS: We'll minimize sequence exchanges, but I'm not ruling them out.

MEMBER HALNON: That would be very helpful, especially with the low number of operators and other things. That all complicates life when you have to do sequence exchange.

MR. HINDS: Sure, sure, and the small, the relatively small number of control rods helps as well, as also the control cell core helps, and the relatively small number of rods.

We also have, I'll mention to you since you mentioned operations, we do have an automation system which helps with control. So, we have capability in the design to automate basically the majority, if not all, of the functions you're alluding to.

MEMBER HALNON: Okay.

MR. HINDS: Okay, next slide?

MEMBER MARTIN: Just real quick, Bob Martin. The TRs that you've presented, is that the extent to which TRs are otherwise incorporated by reference in the CPA? I anticipate that maybe there's a fuel one maybe that follows from other designs that may already have been approved, but --

MR. HINDS: I don't know. Kelli, do you have a comment to that, topical reports incorporated by reference?

MS. BANKS: So, the topical report that is incorporated by reference is NEDC-33922, I think, is the right number, the containment evaluation methodology licensing topical report. That one is incorporated by reference because it, you know, provides summaries of the methodology and things like that.

The remainder of the LTRs are referenced within the PSAR where appropriate. Limitations and conditions, you know, are addressed within the PSAR. The only one that is incorporated by reference is that containment I had mentioned.

MEMBER MARTIN: Okay, well, I wasn't really covering all of that, so, but nothing else, like, related to fuel?

MR. HINDS: Well, as we've said, we're highly leveraging the past history of methods, use of TRACG, for example, that do continue.

MEMBER MARTIN: You mentioned GNF2 fuel and I know there's a topical report, possibly, once upon a time.

MR. HINDS: I'll need to defer to --

MEMBER MARTIN: I think it incorporated it into --

MR. HINDS: -- Ray or Kelli on the regulatory treatment.

MR. SCHIELE: So, there are quite a few topical reports that are referenced.

MR. HINDS: Yeah.

MR. SCHIELE: But in Chapter 1, I believe there's two things listed as incorporated by reference, and that's the containment performance that Kelli mentioned and the QA topical report for 17.5. Those are the only two things we have listed as totally incorporated by reference.

MR. HINDS: Thank you.

MR. SCHIELE: Go ahead.

MR. HINDS: Okay, next slide, please. Okay, so now transitioning off of the LTRs and just going back to the same design features that I introduced before, so just a transition slide. If you could move to the next slide, please.

We've already hit on some of these topics, but I give you little quick visuals as well anyway just to reinforce the history. Natural circulation is known and we do have historical data as well from plant, excuse me, plant operation, as well as from tests, and so this is just a very brief summary of some of the history.

So, I'll highlight a few things on here, one I've already mentioned, but I'll mention again, is that Dodewaard is heavily leveraged in our history and analysis. We also have stability testing that was performed. We have, that was mentioned, operating plant.

Again, that was a forced circulation plant that had a recirc pump trip, and we did stability data gathering, which once the pumps are tripped, it becomes a natural circulator, and so the data is useful for both forced circulation plants as well as natural circulating plants.

We've done chimney two-phase flow testing to ensure that the chimney, which is the area annular space up above the reactor pressure vessel where the steam transitions up to the separators and dryer, the chimney two-phase flow, it's very important that we understand that, and we do, and we've done testing to show that, and our computers models for tests check against the analytical, I'm sorry, the test data.

Start-up characteristics, a natural circulator starts up differently than the forced circulating plants. We've done testing for start-up, and I've already mentioned TRACG is qualified to this type of data.

The part I mentioned on stability, we were very strategic about our selection of core size and also the core power density. So, we've selected core size and power density to make it behave very well, natural circulation in a stable fashion, and we've proven that analytically and it's backed up by these tests. So, much of this, I've already covered, but it's just a reinforcement. Next slide?

CHAIR HARRINGTON: This is Craig, real quick.

MR. HINDS: Yes?

CHAIR HARRINGTON: The chimney, and two-phase flow, and start-up characteristics, were those primarily non-nuclear test facilities or a little of both?

MR. HINDS: Yes, those tests were not using nuclear fuel. Now, of course, that operating BWR stability test was an operating plant, but the other bullets that you're pointing to or referring to, those, there was no nuclear fuel. Of course, we used nuclear-grade quality controls. However, they were not nuclear fuel.

CHAIR HARRINGTON: Okay.

MR. HINDS: But the thermal-hydraulics still stand.

CHAIR HARRINGTON: Sure.

MR. HINDS: Okay, next slide, please. This is just showing the configuration of the reactor pressure vessel and the internals. Just, and some of it is talking about our operating experience to also give a brief summary of the configuration of the reactor pressure vessel.

So, the design and fabrication of the RPV is consistent with our past designs and very well know, so operating experience, as well as design and fabrication experience. There's the KKM plant I mentioned, which we leveraged for the core size and design, the chimney, the Dodewaard plant did have a chimney, and we've done testing to also prove chimney behavior.

So, to make sure everyone knows what I'm talking about with the chimney, maybe you can see it's pointing to the chimney region. It's in the center of the RPV in this figure, and the chimney is simply a cylindrical steel structure on top of the core shroud assembly. The core shroud is underneath and the fuel is within the core shroud. If you can read the labeling, it will help.

The shroud is the cylinder, steel cylinder around the fuel. The chimney is the steel cylinder with just a steam space. The control rods come in from the bottom, and as with our other BWRs, they control reactor power.

So, the left descriptions give just a little more about our history and understanding of these individual components. The steam dryer is virtually the same or very similar to past dryers, and the separators are the same or similar to past steam separators.

The separators and dryers, of course, are in the upper portion of the reactor vessel on top of the chimney drying the steam as it exits. I already mentioned the fuel is widely used and the control rods as well.

The fine motion control rod drives, we have experience from them, from the design and operation of the ABWR and the design of the ESBWR. So, they have operational experience, two means of insertion, and one means of withdrawal. The one means of withdrawal is by motors. The two means of insertion is hydraulics and motors. I have another figure that shows the simple assembly of the fine motion control rod drive. Next slide, please.

CHAIR HARRINGTON: Real quick.

MR. HINDS: Yes?

CHAIR HARRINGTON: This is Craig. The shrouds on there --

MR. HINDS: Yes.

CHAIR HARRINGTON: -- between the core plate and the top guide --

MR. HINDS: Yes.

CHAIR HARRINGTON: -- is there something in there?

MR. HINDS: That's the fuel. That's the core. So, sorry, we didn't -- we only have one, it looks like one bundle in this visual here, but the core is inside the shroud.

So, the core -- the fuel, the 240 fuel bundles fit inside that area above the core plate and supported on the bottom by fuel support castings, which are inserted into the core plate, and the top, you get lateral support there from the top guide.

CHAIR HARRINGTON: I failed to read the label of control rod guide tubes at the bottom and was assuming that was the core.

MR. HINDS: It's easy -- it's a lot of stack-up here. So, the control guide tubes are down there now as you see, and then the control rod drives down below.

CHAIR HARRINGTON: That makes entirely --

(Simultaneous speaking.)

CHAIR HARRINGTON: I just didn't study it close enough.

MR. HINDS: Okay? And again, that's just a steam space up above the core. So, this would be very similar to an existing boiling water reactor except for we insert the chimney in there, and there's an annular space around both the chimney as well as the shroud, which is where the downflow comes for natural circulation. Okay, next slide, please.

This is just a representation of the fine motion control rod drives, and again, there's motors and there's hydraulics, and the motors provide for insert and withdrawal. The hydraulics will lift if needed, and if a hydraulic scram occurs, lift the control rod basically up off of the ball nut and insert it.

And so, these two means, although they move the same control rods, they can function independently. So, a scram from hydraulics is not impacted by the motors, and the motors can insert regardless of whether the scram worked or not, so two means of shutdown of inserting control rods. Normal power control is with the motors, with our fine motion and very small movements. Next slide, please.

This is showing the core representation and some instrumentation. We have in-core instrumentation similar or basically the same as our operating experience with local power range monitors or LPRMs. They're inserted within the core and spaced around to their appropriate locations within the core such that we get a full power reading at power, during power operation.

And then for startup and low-power operation, we also have, within the core, wide-range neutron monitors. All of those neutron monitors, they are non-movable. They're fixed in-core. We calibrate our local power range monitors by using gamma thermometers that are integral on the LPRM string.

They sense the gamma flux and equate that to a neutron flux and use that as a calibration standard for the local power range monitors. We also have a small little representation out on the periphery showing of an in-core water level measurement from below.

CHAIR HARRINGTON: So, two questions. As a PWR guy, the neutron monitors, do they also provide flux mapping capability axially or --

MR. HINDS: Yes, the LPRMs have sensors that are spread in a predetermined fashion on the axial, as well as we have them placed in the designated locations you can see on the figure here radially, but yes, there's four sensors there giving power levels at different axial locations.

And then the gamma thermometers are there to actually give a diverse indication of the neutron flux that can be used or the gamma flux that can be correlated to the neutron flux and that then can be used to compare for calibration standard, but yes, there is an axial measurement.

CHAIR HARRINGTON: And then the water level, water sensing, what type of sensor?

MR. HINDS: That's a heated junction thermocouple type, yeah, and it's just a diverse means of water level sensing. It's only in the worst case, if water level were to be extremely low. Our normal water level sensing should sense normal operation as well as the majority of accidents. For a very extreme low-frequency accident, if the water level were to get very low, this is the backup.

CHAIR HARRINGTON: Thank you.

MR. HINDS: Next slide, please.

MEMBER ROBERTS: I'm not very familiar with this technology. How do you correlate the gamma flux to power?

MR. HINDS: We've done in-core testing, operating reactors, and we've done a significant amount of analysis. There was quite a number of submittals to the NRC under the ESBWR where that technology was covered in licensing space, but there's a significant amount of testing, both up to and including in operating reactors, and the gamma flux does have, we have a correlation of gamma flux to neutron flux which can be used to correlate to actual reactor power.

MEMBER ROBERTS: So, there's no calorimetric calibration required?

MR. HINDS: We do have a heat balance that's running live time at all times. So, we have a heat balance that's checking and used to calibrate for the gain adjustment factors for the neutron monitors.

So, the gamma thermometers are primarily there to -- they're local. The heat balance is global or the whole system. So, the heat balance for the whole system, the gamma thermometers can give us a flux distribution across the various locations, both radial and axial, within the core.

So, we can take that heat balance and correlate that to the total core power, and then we can use the gamma thermometers to assign that to the various locations within the radial and axial locations within the core, so we do both, but yes, there is a, quote-unquote, calorimetric. We call it heat balance.

MEMBER ROBERTS: Okay, that makes sense. Thank you.

MR. HINDS: Okay, I've mentioned several times the reactor isolation valves. Here's a visual representation, dual valves attached directly to the reactor pressure vessel and supports the coolant preservation approach.

There is no piping between the reactor isolation valves and the reactor vessel, so thereby, there is no pipe that could break inboard of the reactor isolation valves.

The nozzles associated with the reactor pressure vessel are forged nozzles. They're integrally fabricated with the reactor pressure vessel, and then very significant bolting is used, meaning, when I say significant, very high-strength bolting used to connect a dual valve assembly directly to the reactor pressure vessel, thereby supporting the coolant preservation approach. So, if there were some threat to coolant loss in the connective system, these two redundant valves would close to isolate the associated system.

MEMBER ROBERTS: Can you explain what large means in that first bullet? So, that implies there are some penetrations that are so --

MR. HINDS: Yes.

MEMBER ROBERTS: -- large they don't have valves?

MR. HINDS: Excellent question. Yes, so we've characterized every reactor vessel nozzle as whether it needs the reactor isolation valve or not. The only ones without reactor isolation valves are very small instrument line connections for reactor water level and pressure sensing.

And those are very carefully located height-wise. They're at least four meters above the top of active fuel, and they're also sized such that even if we were to have a double-ended break of those sensing lines, we would still maintain fuel cooling over an extended period of time.

And within the PSAR, within the construction permit application, we present an analysis to show the response of the plant if we were to have a break of those sensing lines, but anything bigger than those sensing lines, which are less than an inch in size, they would have integral isolation valves.

MEMBER ROBERTS: So, there's some sort of makeup capability that can keep up with a small line break?

MR. HINDS: Well, as I mentioned in some of the lead-in of the coolant preservation approach, we start with a lot of coolant in this plant. Being a natural circulator, we can break one of those instrument lines, have zero makeup, provide decay heat removal with the isolation condenser system.

There's enough coolant to last in excess of three days. So, the passive capability requirement of three days' cooling can be maintained, zero makeup, even in the presence of an instrument line break, and that's demonstrated by analysis in the PSAR.

MEMBER ROBERTS: Within the three days, you'd have the capability to --

(Simultaneous speaking.)

MR. HINDS: Yes, we do have -- so, the control rod drive system that was shown with the control rod drives in the bottom of the vessel has a normally running, at-all-times running purge water system function to keep the control rod drives nice and clean, and that purge water does serve a dual function as a makeup.

Not much water is needed. Those instrument lines, the coolant loss is minimal, especially with the isolation condenser system providing the cool down and depressurization in those types of events.

MEMBER ROBERTS: Okay, thank you.

CHAIR HARRINGTON: And I read someplace, I think, that you also credit those, that kind of a leak as helping to achieve pressure balance between containment and the vessel or --

MR. HINDS: We don't --

CHAIR HARRINGTON: -- am I making that up?

MR. HINDS: We don't -- I understand the topic, I think, that you're touching on. The containment pressurization actually can help to limit any potential coolant loss. I think that's what you're referring to, is --

CHAIR HARRINGTON: Yeah.

MR. HINDS: -- meaning we would catch and collect that coolant if it were to be lost, say, for instance, in an instrument line break or other line breaks.

As the containment pressurizes and as the reactor depressurizes, the equalization serves to limit the coolant loss, and the isolation condenser system capacity is large enough that it does depressurize the reactor.

CHAIR HARRINGTON: And so, between the two isolation valves, I would assume that trapped volume is ported back to the vessel side somehow if it, so that it doesn't heat up and --

MR. HINDS: We do evaluate the potential for pressure lock or any of those types of functions, but it's not ported back to the reactor vessel, and that is a dual-valve assembly that we're showing here.

We're running either out or very short of time, so you can stop me at any moment, but I'll keep on moving until the point where you would like me to stop. Next slide, please.

MEMBER MARTIN: I just can't help myself, sorry, Craig. What's the technology maturity of those IAVs or RIVs? Have you all fabricated and done some testing or are you still on paper?

MR. HINDS: So, the testing, physical testing of our specific valves has not yet occurred, but is planned. We are using, I'll say -- I'll avoid mentioning company names right now, but we are using well-known, reputable valve and actuator suppliers such that it's not their first introduction of valves.

We're leveraging their technology evolution and basis, similar to what we were leveraging our own technology evolution such that we have high confidence in the concepts and the specific application, but the physical specific tests that we've asked them for has not yet been performed.

Because we've asked them to perform steam shutoff tests to prove that, with our conditions, in a worst-case type coolant loss, that the valves can close and --

(Simultaneous speaking.)

MEMBER MARTIN: But ultimately, it's also trying to eliminate breaks at those locations, right?

MR. HINDS: The, again, the --

MEMBER MARTIN: Well, in, say, a LOCA, eventual LOCA analysis, will you still --

MR. HINDS: Just to clarify, there is no piping, or to repeat, there is no piping inboard of the reactor isolation valves --

MEMBER MARTIN: Right.

MR. HINDS: -- and that is a reactor pressure vessel nozzle that's directly attached to a valve, and that nozzle is very robust, similar, basically thicker than the reactor pressure vessel by virtue of the shape --

MEMBER MARTIN: Right.

MR. HINDS: -- and fabricated to the same standards of the reactor pressure vessel, and the bolting is quite significant.

MEMBER MARTIN: Okay, I just wanted to make sure.

MR. HINDS: There are --

(Simultaneous speaking.)

MEMBER MARTIN: -- record, really, or you hadn't said it yet.

MR. HINDS: There are no piping welds inboard.

MEMBER MARTIN: Are these --

MR. HINDS: Oh, and there is forged assembly, and it's a forged valve assembly, forged nozzle assembly and forged valve assembly directly bolted together. The only weld for the nozzle, the nozzle is actually part of the reactor pressure vessel fabrication, which, you know, reactor pressure vessels have full penetration welds.

MEMBER MARTIN: Sure.

MR. HINDS: Those nozzles are part of the reactor pressure vessel fabrication.

MEMBER HALNON: Is this a feature of the Darlington reactors?

MR. HINDS: It is.

MEMBER MARTIN: Okay, that was my question, whether -- okay.

MR. HINDS: Yes, okay, next slide, please. Okay, this is somewhat of a repeat, and the only reason I included it is just to trace out one system, this is the main steam system, to show you how it's configured as a system.

So, you can see the reactor isolation valves directly attach. We have two steam lines directly attached to the reactor pressure vessel. Then the piping runs out of containment. The containment penetration is shown as that sleeve area, and then there's additionally an outside containment isolation valve for the main steam system.

So, there's three valves in series where historically we've had two, and historically they've been further removed from the energy source. So, now we're having the ability to shut off right at the energy source. That's our strategy there, coolant preservation, and the isolation condenser systems help to enable that to occur because the cooling is maintained. Next slide, please.

You can stop me any time you're ready to move on, but this is the isolation condenser system, which I've mentioned several times. We have -- we meet the passive plant rules of in excess of three days of cooling of our passive system, but we, instead of having a makeup to get between three and seven days, we have already sized the cooling such that we have in excess of seven days' worth of heat removal contained in these pools.

And Ray had already told you about the pool figuration. There's a pool on one side of the structure, with the Alpha heat exchanger immersed in it, and there's a pool on the other side with the Bravo and Charlie heat exchangers immersed in it.

So, there's three trains of isolation condensers, and all that's needed to place them in service -- you can see the blue valves down towards the lower portion of the figure. There's two parallel redundant valves. They're condensate return valves.

All that has to happen is one of those two valves has to open. They're configured to fail open. So, the failsafe nature here is this system fails in a cooling function. Those valves, if they lose either a mode of force or electrical power signal, they would fail open and the system would go into service, similar to all our other safety functions, fail in a scram.

So, the failsafe nature of this plant is that it would fail with the reactor shutdown via hydraulic scram with stored energy. The reactor would isolate via the reactor isolation valves.

The isolation condenser system would be placed in service by simply opening one or two, or both of those valves, allowing condensate flow to come back from the heat exchanger. All it is, is steam flows in from the reactor into the heat exchanger. It's condensed.

The LEGO (phonetic) condensate comes back to the reactor vessel and it flows. We've chosen to pour it into the chimney region of the reactor. It helps to suppress pressure very well by putting it in the chimney region. It actually suppresses core flow and helps in even more severe events.

MEMBER HALNON: Is each ICS 100-percent duty?

MR. HINDS: Yes, so for --

MEMBER HALNON: How are you going to swing -- I mean, the Charlie one is sort of an extra?

MR. HINDS: We have a consideration of a rotation, if you will, for -- I think you were talking about duty on them. In a transient, all we would need is one, and in a pressurization transient, all we would need is one.

In a LOCA, we've chosen to configure the logic such that all three will initiate. So, all three will initiate, and that provides the maximum cooling, and therefore depressurization in a LOCA to help minimize coolant loss. I'll keep moving and stop at any time.

CHAIR HARRINGTON: And we're going to try to get through all of these slides, but do you have a concern about overcooling since you've got 300 percent?

MR. HINDS: Excellent question. So, yeah, there was some -- we looked very hard at did we over-design and oversize? And, you know, it's very, very beneficial to have them large from the standpoint of pressure suppression, over-pressure control, depressurization of LOCA.

And, oh, by the way, these have undergone full-scale testing of this specific design as part of our evolutionary design for SBWR and ESBWR. So, we wanted to preserve that testing and we also wanted the excess capacity.

So, to your question, part of our design of the reactor pressure vessel and the isolation condenser system is to build into it the thermal cycles and the stress associated with the cool-down effects of both transients as well as accidents.

We select the numbers of postulation through postulation of how many we would put into the design of those metal components to build those stress cycles, so it's already built-in, the stress cycles, into the nozzles, the RPV, the system, so we have built in the cooling effect.

Now, of course, we select the cycles appropriately based upon the probability of a LOCA and how many the plant would be postulated to have in the life of the plant.

CHAIR HARRINGTON: And reactivity as well, I assume, is not a concern for that kind of --

MR. HINDS: Oh, no. Now, I mentioned briefly that it discharges into the chimney area. Discharging into the chimney area mitigates the reactivity portion since we're not shooting cold water directly into the inlet of the reactor, so it actually helps to suppress power.

So, if we were to have a fail-to-scream, initiation of the isolation condenser system helps with both pressure control and reactivity control, as opposed to herding reactivity control. Good question.

MEMBER HALNON: Seven days' coping time is -- days four through seven, any operator action at all?

MR. HINDS: No operation action.

MEMBER HALNON: So, it's fully happening --

MR. HINDS: Yes.

MEMBER HALNON: -- all the way through?

MEMBER MARTIN: The other thing about the seven-day, I assume that's assuming all your uncertainties, your conservative, your safety analysis. Realistically --

MR. HINDS: We can go a long time beyond that.

MEMBER MARTIN: I mean, have you -- you know, of course, you have 300-percent capacity to cool, but is your pool sized? I mean, is it --

MR. HINDS: Yes.

MEMBER MARTIN: -- really 21 days, you know what I mean, or --

MR. HINDS: I won't commit to you all how many days beyond seven, but it's beyond seven. There is some detailing of some of the, you know, overflows, drains, and all of the details of the pools that can have impacts to that.

(Simultaneous speaking.)

MEMBER MARTIN: -- connectors between the pools, right?

MR. HINDS: Yes. Oh, yeah, let me describe that a little further, and we significantly will exceed seven days, so, to your general point, but the pool, the inner and the outer pools. The inner pools are segregated such there's no communication out of the inner pools. You cannot lose water from the inner pools even if you lose the entire outer pools.

The outer pools serve -- they are interconnected. The outer pool is basically one functionally, and the outer pool provides passive flow into the inner pool, and it's via underwater piping and check vales, so one-way flow, the water from the outer to the inner, but it cannot come out of the inner, regardless of what you do to the outer pool.

And so, the outer can makeup to any of the inner pool, both of the inner pools, and the inners are segregated such that the Bravo Charlie does not communicate at all with the Alpha pool.

MEMBER MARTIN: So, the inner pool, just so I'm 100 percent, the outer pools will cover the other maybe 200 pools.

(Simultaneous speaking.)

MR. HINDS: Yeah, you know, for the seven-day coping, we're going to need the outer pool.

MEMBER MARTIN: Okay.

MR. HINDS: But the inner pool is primarily there just to protect the first few days.

MEMBER MARTIN: Sure.

CHAIR HARRINGTON: Is there some logic or issue for having two trains in one pool?

MR. HINDS: It's mostly geography within the building. Yeah, and so what we wanted to do was have the nozzles coming off the reactor pressure vessel around, you know, and segregate the trains within the containment, and we wanted to shoot as close to straight out.

We want to minimize crossties and things such as running all the way around containment, and so we go almost straight up, so it's mostly geography of the piping runs.

CHAIR HARRINGTON: And you had a quadrant for access outside for casks, and fuel, and all that?

MR. HINDS: Yes, so the center between them is used primarily for refueling activities. So, there needs to be a segregation of one side of the building to the other basically primarily for the reactor cavity and the refueling.

CHAIR HARRINGTON: And one side had more real estate to fuel and the other side --

MR. HINDS: Yes.

CHAIR HARRINGTON: -- needed some real estate?

MR. HINDS: Yes, because as you noted, there is, in some quadrant, there is accesses with no water, so, yes, you're correct. Next slide, please.

This is just more of the same. I think I've covered everything here. It's just a different figure to show more functionally with the valves.

But again, I'll reinforce that this system is always pressurized, steam pressure on top and then condensate back on the backside, and the only thing preventing it from actually flowing is one of the two, the two closed condensate return valves that are configured such that they'll fail open.

MEMBER ROBERTS: So, what is the strategy for containment isolation, your third bullet there? We talked about it earlier this morning. There's a conflicting safety requirement.

MR. HINDS: Yeah.

MEMBER ROBERTS: You want to have the system online for cooling, but you need to sometimes isolate it for containment.

MR. HINDS: Yes, we've prioritized this system and we designed -- first off, we designed from, like, our Defense Line 1 type of approach. We've given the highest ASME class, the highest safety class, and the highest treatment of this system to ensure that it will not break, and stress rules, et cetera.

This is an ASME Class 1 system in its

entirety. It's considered part of the reactor coolant system boundary. The valves for isolation are all attached directly to the reactor pressure vessel, those reactor isolation valves. There is no valve outside of containment for isolation.

It's a closed-loop, a simple closed-loop that goes in the heat exchanger and comes immediately right back. The prioritization within our logic and control scheme is that cooling wins. The cooling function is the safety function of this system.

The likelihood that we were to have a coolant loss from this system is very, very low. The likelihood that we would need this system for cooling is significantly higher. The safety function is the high-priority function. We do not ignore the potential for loss of coolant, and we do have leak detection and isolation, but we do prioritize the cooling function.

Next slide, please. This is just showing a little more on the containment. I've already covered most of this, but it's an evolutionary containment design.

But I'll note that the choice for BWRX-300 was dry containment as opposed to pressure suppression wet containment. And our containment LTRs that we've presented to the NRC and already reviewed presented both the analytical methods, the safety performance. This works very well in conjunction with our coolant preservation approach.

We don't have valves that blow the system down to -- we don't have SRVs to a suppression pool is really what I was alluding to. We have the isolation condenser system rather that's discharging its heat through the closed loop up in the isolation condenser system pool.

So, dry containment was chosen and it's a 60 psi design pressure containment, cylindrical containment, and we built upon both the structural as well as the analytical and safety learnings from the past. Next slide?

This is -- I've mentioned we have a passive containment cooling system. This is it. It's quite simple. It's an array of piping, and the array of piping simply takes water from the pool up above. Now, this is leveraging other pools.

I mentioned briefly that, you know, when we were talking about the geography of the pools, on one side of the building is the Bravo Charlie pool and the other side of the building is Alpha pool for isolation condensers. In the center alley between there is an equipment pool, and a reactor cavity pool, and a fuel pool.

There's gates that segregate the fuel pool from those other pools, but those other pools are always flooded during operation. We've leveraged the water within them for a dual purpose for -- they're used for refueling, but we also use that same water as the heat sink for the passive containment cooling system.

It's very simple. There's no moving parts needed to place this in service. It's always in service. It's just its flow is determined by the heat demand, so just the differential density drives water down the cold leg and runs it back up the hot leg, and so the differential head and differential density causes water movement there, so it acts similar to a radiator, but it's a piping array that's spaced around inside the containment, up very close to the containment inner surface.

It removes heat. It's very, very effective as a condensing surface for steam. If there were to be a loss-of-coolant accident, the steam would condense on these tubes and help to suppress the pressure within the containment.

MEMBER HALNON: I know you've got seismic isolated, but are you concerned, anything about sloshing? You've got these pools that are at the highest point of their structure.

MR. HINDS: Well, they're actually not at the highest point of the structure, but they are up there. They're up --

(Simultaneous speaking.)

MR. HINDS: They're actually right around grade elevation. So, they are, relative to the base mat, they're pretty high, but relative to the ground, they're not high, so that's the kind of --

MEMBER HALNON: Yeah.

MR. HINDS: But, yes, and, of course, we look at any dynamic loads as well.

CHAIR HARRINGTON: And I suppose since all three of those heat exchanger panels lead into the same pool --

MR. HINDS: They do.

CHAIR HARRINGTON: -- the single pool is somehow treated in the safety case as, I guess, it's further down the list of concerns, so having them all in a common pool is not --

MR. HINDS: Yeah, it's just a -- treat it as a common heat sink, but yes, you're correct, and we haven't found failure modes that would cause concern there. It is a very simple system and it doesn't have moving parts. And anyway, we did not see the need for creating a segregation scheme in there, and we do multipurpose these pools.

CHAIR HARRINGTON: And you've got 15 seconds to stop the LOCA in progress so you don't have a huge heat dump to containment.

MR. HINDS: The heat load on these is not -- the heat load on the isolation condenser systems is relatively large. The heat load on these is not anywhere even close. We don't even boil these pools. We boil the isolation condenser pools, but these do help to minimize the peak pressure within the containment, but the majority in a LOCA --

Remember, when an isolation condenser system is initiated, it's depressurizing the reactor. That's where the energy's coming from, and if we're in a LOCA, you know, theoretically, we're communicating -- if we haven't isolated it, we're communicating with the containment, so this is taking what got discharged to the containment and just helping to minimize the pressurization.

We do -- you know, it's a 60 psi containment, so we do have pressurization, but this helps to limit the peak, especially in a small break

LOCA like the one that we were talking about for the instrument line break, you know, where we postulate a continual steam discharge. It's condensing on these tubes.

I think this was the last slide. Is there another slide? That's the last one. Sorry for my time management. I went over.

PARTICIPANT: No, it's completely our fault.

CHAIR HARRINGTON: And this was -- the point of today is yeah, yeah, to cover all of this information, and I think it's been very helpful. Many of us are new on the committee since a detailed presentation from GE on this design, so that was getting us all kind of up to the same point was really the intent today --

MR. HINDS: Thank you.

CHAIR HARRINGTON: -- and we very much appreciate that. Are there other questions?

MEMBER HALNON: Real briefly, from a multi-reactor site perspective, have you envisioned what systems would be shared and economies of scale? How would that look?

MR. HINDS: The current design as developed -- that's a very good question. You know, of course, things could change with time, but the current concept is these, this standard plan is standalone, so it doesn't matter if it's single unit or multi-unit.

Where the sharing is primarily implemented to be design site specific is when you get into support functions around the power block structures. So, the current approach is to make the power block stand alone, and then as desired by individual customers, optimize sharing of administration --

(Simultaneous speaking.)

MR. HINDS: -- security administration, security, even, you know, replenishment or makeup, you know, fire protection.

MEMBER HALNON: What about diesel generators? Do you think that -- is that --

MR. HINDS: Currently not shared. That's an excellent question, but currently there are two diesel generators as part of this standard design dedicated to that unit. That was a very reasonable question.

MEMBER HALNON: Is it designed to be an energy island to where it could be disconnected from the grid?

MR. HINDS: That would be a site-specific customization. The standard design is neutral on that. Now, I will say a little bit to the point you were alluding to. The standard design is not, does not have the 100-percent bypass capability of steam bypass to the condenser.

Therefore, if there's a load reject from 100-percent power, the standard design has a reactor scram because there's, you know, the load reject. We have, within our experience base, certainly the capability to introduce a higher bypass capacity, but the current standard design does not have 100-percent bypass.

MEMBER HALNON: It's not a black start, is it?

MR. HINDS: Not part of the --

(Simultaneous speaking.)

MEMBER HALNON: Not the diesel generators, I mean.

MR. HINDS: Not part of the standard design. We would need a power source to power features such as condenser cooling water or circulating water.

The diesel generators back with the standard design are sized to back functions such as fuel pool cooling, shutdown cooling, but more nuclear-specific functions, from a defense-in-depth, and restoration of battery chargers and recharge your batteries.

This is all sized for that, but not sized large enough to power and configure it in the bus work to power the circulating condenser cooling water or cooling towards, for example, so additional power would be needed for, you know, for example, cooling tower loads.

MEMBER HALNON: Last question, extreme temperatures, how does it look for very hot and very cold?

MR. HINDS: I can't recall the TVA PSAR temperature extreme limit, but we do have quite a range of temperatures from cold, very cold to very hot, you know.

MEMBER HALNON: I mean, you're putting it in Canada, so.

MR. HINDS: Yes, you know, working in a snowbound territory.

MEMBER HALNON: Just so you know more about the cold, I mean --

MR. HINDS: Yeah.

MEMBER HALNON: -- they have a pod.

MR. HINDS: Yes, we do have consideration for adjustments primarily within the HVAC system on a site-specific basis because it doesn't make great sense to have all of the HVAC sizing completely standard, so there is allowance for site-specific adaptation to, you know, resizing of some of the HVAC components. Their structures are built to accommodate that.

MEMBER HALNON: Thanks.

MEMBER MARTIN: I'll just follow that up. See, you do have a specific design. These guys have a specific design in mind. Are you all just looking to deploy the standard plant or have you asked for anything unique that might touch on some of the things Greg said?

MR. HUNNEWELL: So, we have not started the site-specific design. We have considered things such as islanding and black start capabilities, and that would be really driven by if there was a need. For example, we're adjacent to the Oak Ridge facility.

If Oak Ridge came along and said, hey, we want you to be part of our resilient power supply and you need to have black start capability, that's when we would likely look at that, because it does add costs and we are very cost conscious on it.

MEMBER MARTIN: Yeah, I know, of course,

Clinch River has been thought of in the sense of supporting Oak Ridge for a long time specifically, so you can see them having unique needs.

MEMBER HALNON: Yeah, and you've also got Hermes up there too.

MEMBER MARTIN: And then you have Hermes, so, yeah, a little competition for small things there, huh?

MR. HUNNEWELL: Very small.

CHAIR HARRINGTON: Other questions? Do any ACRS members or consultants online have questions?

MEMBER KIRCHNER: Yes, I have one, Craig. This is Walt Kirchner. Thank you for the presentation, everyone. Just, it was mentioned that when the containment, when the isolation condenser system is operated, that would probably lead to boiling.

Would the normal configuration be to close the reactor building, isolate that as well, and would that then contribute to a very wet atmosphere inside the building in terms of the equipment qualification, et cetera?

MR. HINDS: For the isolation condenser system, as I mentioned, there's a roof, if you will, or there's a slab up above the isolation condenser pools.

MEMBER KIRCHNER: Right.

MR. HINDS: And that slab serves as, you know, a top for the pools, and there is a vapor space between the top of the pool surface and the slab. That vapor space is vented outside. So, there is a --

MEMBER KIRCHNER: That's vented outside, okay.

MR. HINDS: Yes.

MEMBER KIRCHNER: Can you isolate that if necessary?

MR. HINDS: It's --

MEMBER KIRCHNER: Or you would isolate a condenser, one of the three trains if you, for whatever reason, detected a release?

MR. HINDS: Yes, the latter, what you just said, in that --

MEMBER KIRCHNER: Okay, okay.

MR. HINDS: -- we do not isolate them. We do not isolate the vent because that would basically plug up the, you know, basically tend to pressurize the pool vapor area --

MEMBER KIRCHNER: Sure.

MR. HINDS: -- but we do have the leak detection and isolation for, if there was a leak within a train, to isolate that train and that train only.

MEMBER KIRCHNER: Okay, thank you.

CHAIR HARRINGTON: I noticed in the PSAR that it mentions that you are not distinguishing between identified and unidentified leakage? I don't know. I'm a PWR guy. I don't know if that's typically in a BWR or is new here. Can you speak to that?

MR. HINDS: So, on past BWRs, say current forced circulation BWRs, for example, like Browns Ferry, for example, there is reactor recirculation pumps. They took the green, what is it, green side there, you know, so that's a forced circulation plant, reactor recirculation pumps. Those recirculation pumps have a designed seal leak-off. That designed seal leak-off is routed to an equipment train tank within that design.

It's planned to have flow there, and because there's planned flow, then there is the segregation for things that were to leak in the containment, so you could segregate the two. In this plant, we do not have design leak-offs such as that within the containment.

So, anything that were to be, call them

present coolant, collect in the sump, it would be from something that's not planned. It would be a leaking component, for example, so that's the unidentified leakage.

So, because there was no designed leak-off, we did not include the equipment trains. They're all just unidentified and collected in a common sump.

CHAIR HARRINGTON: And not a lot of other water sources in containment that would be confused without any --

MR. HINDS: No, it's --

CHAIR HARRINGTON: Okay.

MR. HINDS: There are coolers within containment, but we also collect the condensate from the coolers, which would be indicative. If there's condensate in the dry containment, then it must have been some vapor coming out of some component, for example.

CHAIR HARRINGTON: Okay, all right, that all makes sense. Thanks. Any other last questions? Okay, well, we very much appreciate your time today to prepare and come. As we -- in the coming months, we'll be starting our review and trying to figure out how to focus that. I mean, this session today is very helpful.

MEMBER HALNON: Public comments?

MR. HINDS: Yeah, we do have to do that, appreciate that. So, anything else before we go out for public comments?

MR. NGUYEN: Yes, I was going to say public comments.

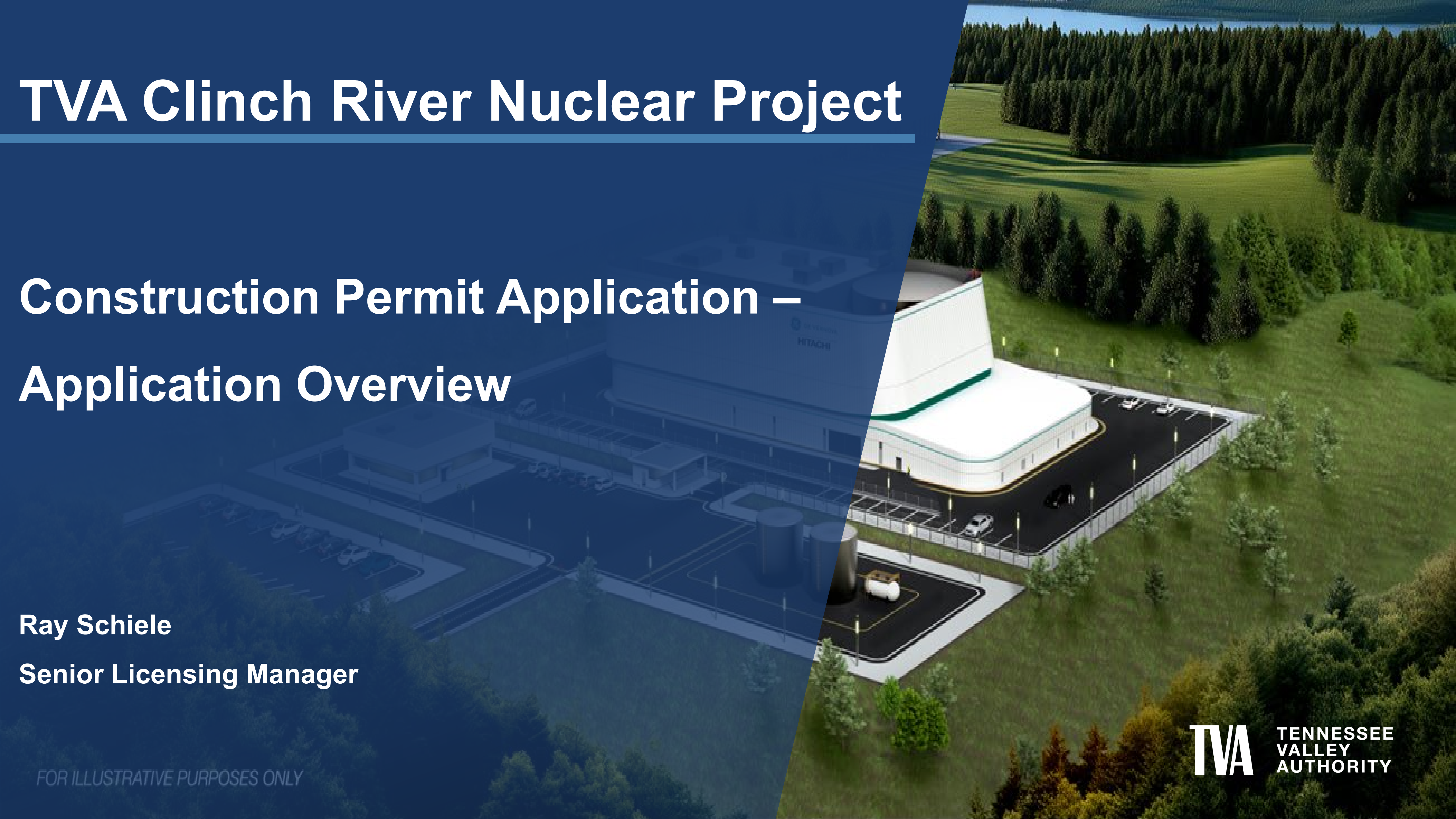
CHAIR HARRINGTON: Okay, if there is anyone online in public that has a comment to make yourself, then do so. Identify yourself and your, any organizational affiliation, and make your comment.

I don't see any indication of any public comments, so I think, with that, we can adjourn the meeting. Thank you again very much for coming today. With that, the meeting is adjourned.

(Whereupon, the above-entitled matter went off the record at 11:46 a.m.)

TVA Clinch River Nuclear Project

Construction Permit Application – Application Overview



Ray Schiele

Senior Licensing Manager

TVA TENNESSEE VALLEY AUTHORITY

Topics

Introductions

TVA Mission and the Role of New Reactors

Pathway from Early Site Permit to Construction Permit Application

Structure/Content of the Construction Permit Application

Questions

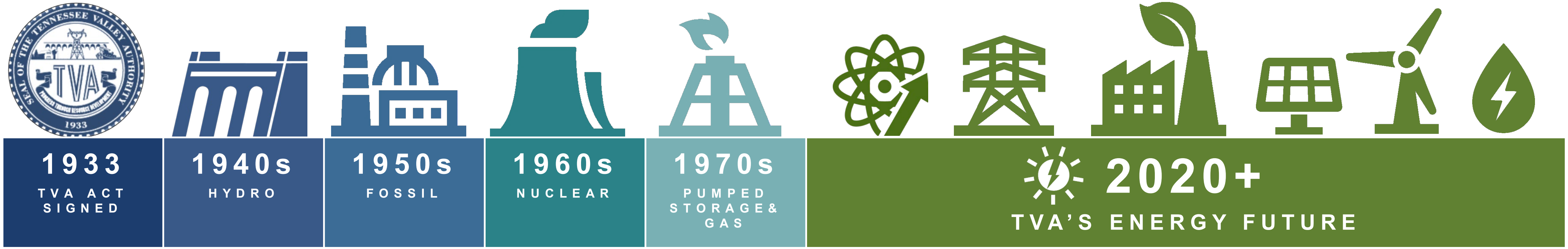
TVA Mission and the Role of New Reactors

BUILT FOR THE PEOPLE OF THE VALLEY

ENERGY — Electricity at the lowest feasible rate and highest feasible reliability

ENVIRONMENT — Stewardship of the natural resources for best use by the public

ECONOMIC DEVELOPMENT — To attract and retain good jobs and capital investment in the Valley



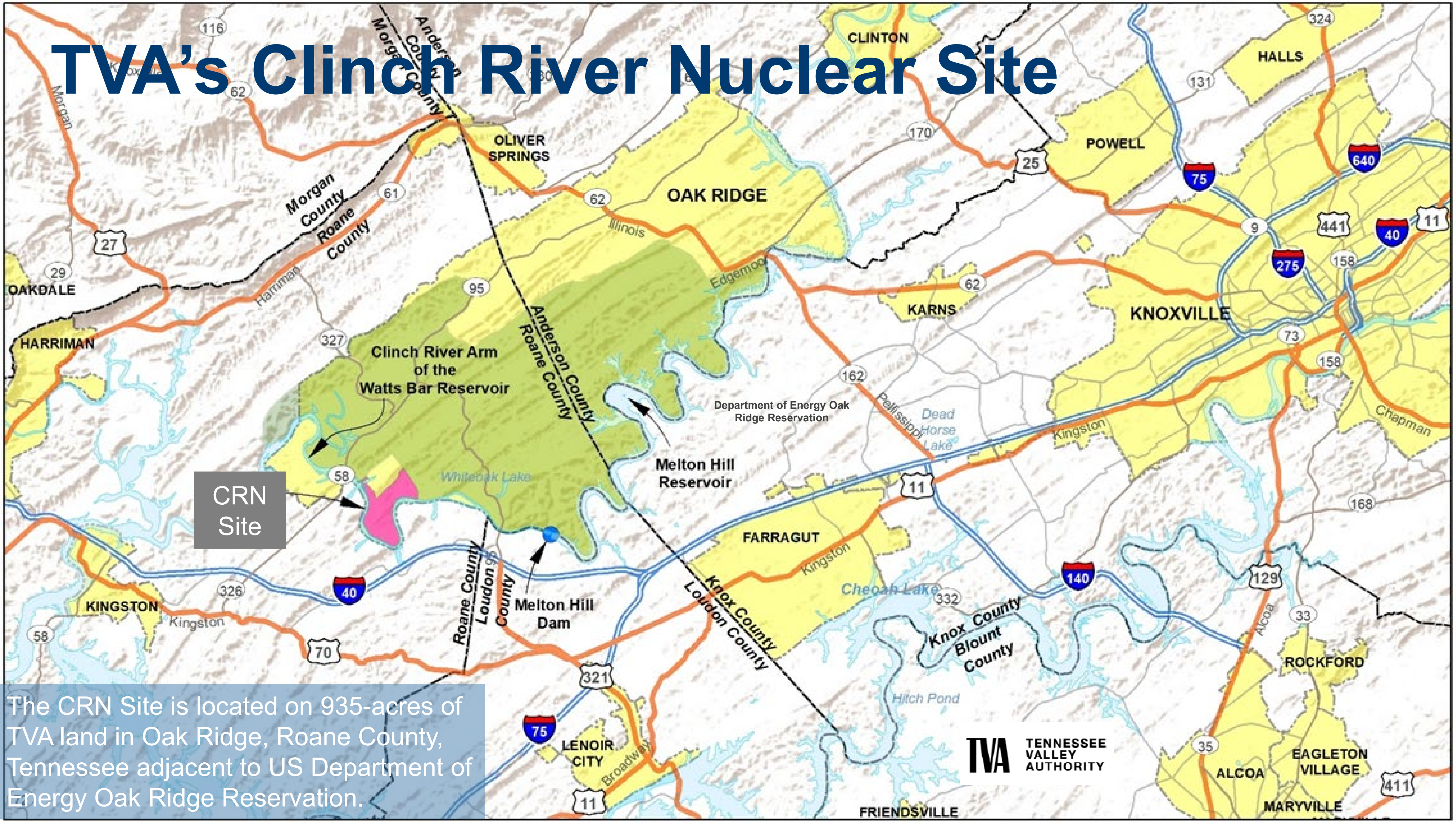
Since its inception, TVA has innovated to meet the needs of the Valley.

Today and in the future, the Valley needs **affordable, reliable, resilient, and secure energy** to **lead the nation in energy innovation and economic development**.

INNOVATING FOR THE PEOPLE OF THE VALLEY

TVA TENNESSEE VALLEY AUTHORITY

TVA's Clinch River Nuclear Site



The CRN Site is located on 935-acres of TVA land in Oak Ridge, Roane County, Tennessee adjacent to US Department of Energy Oak Ridge Reservation.

TVA's ESP to CPA Submittal Timeline

2019 | 2020 | 2021 | 2022 | 2023 | 2024 | 2025

ESP Issued 12/2019

ESP Issued

SMR Technology Evaluation

- LCOE
- Supply Chain Constraints
- Advanced Manufacturing
- Seismic Considerations
- Modular Construction
- Advance Construction Techniques

CPA Scoping

Annotated Outlines:

PSAR

- RG 1.70
- NUREG 0800

Environmental Report

- RG 4.2
- NUREG 1555

CPA Development

- Topical Report for NQA-1 CRN 1 Program
- Exemption Request 2.101(a)(5)
- Exemption Request Excavation
- TVA NEPA and Permitting

- Select GVH BWRX-300 Technology
- Development of Regulatory Framework Documents
- Develop an Environmental Report for CRN-1
- Develop 10 CFR 50 Construction Permit Application
- LIC 116 Readiness Assessment

CPA Submittal

CPA 2 Parts:

- Part 1 ER 4/2025
- Part 2 PSAR 5/2025

CPA Acceptance Review

- ER 6/12/2025
- PSAR 7/9/2025

CPA Audit

- ER 7/14/2025
- PSAR 7/14/2025

Construction Permit Application Content (CPA)

Content of TVA CRN-1 CPA

Enclosure 1- General and Administrative Information

- 10 CFR 50.33 Contents of applications; general information

Enclosure 2 – Preliminary Safety Analysis Report [Non- Public]

- 10 CFR 50.34(a) Contents of applications; technical information.
- NUREG 0800 Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: (LWR Edition)
- Reg Guide 1.70 Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)

Enclosure 3 – Preliminary Safety Analysis Report [Public]

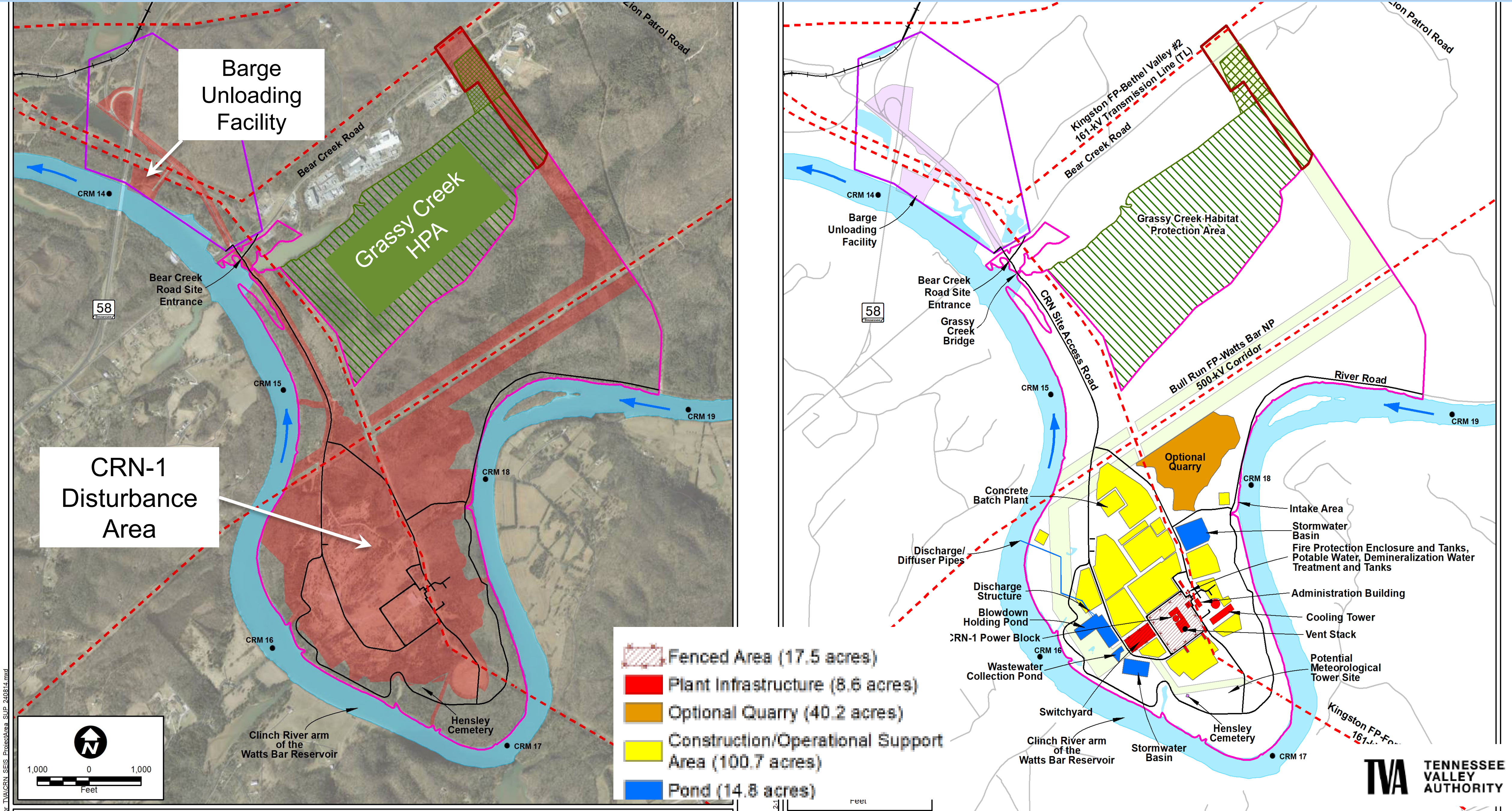
Enclosure 4 – Exemptions and Variances

- 10 CFR 50.12 Specific exemptions
- 10 CFR 52.39 Finality of early site permit determinations

Enclosure 5 – Environmental Report

- 10 CFR 51.50 Environmental report-construction permit, early site permit, or combined license stage
- NUREG 1555 Standard Review Plans for Environmental Reviews for Nuclear Power Plants
- Reg Guide 4.2 Preparation Of Environmental Reports For Nuclear Power Stations

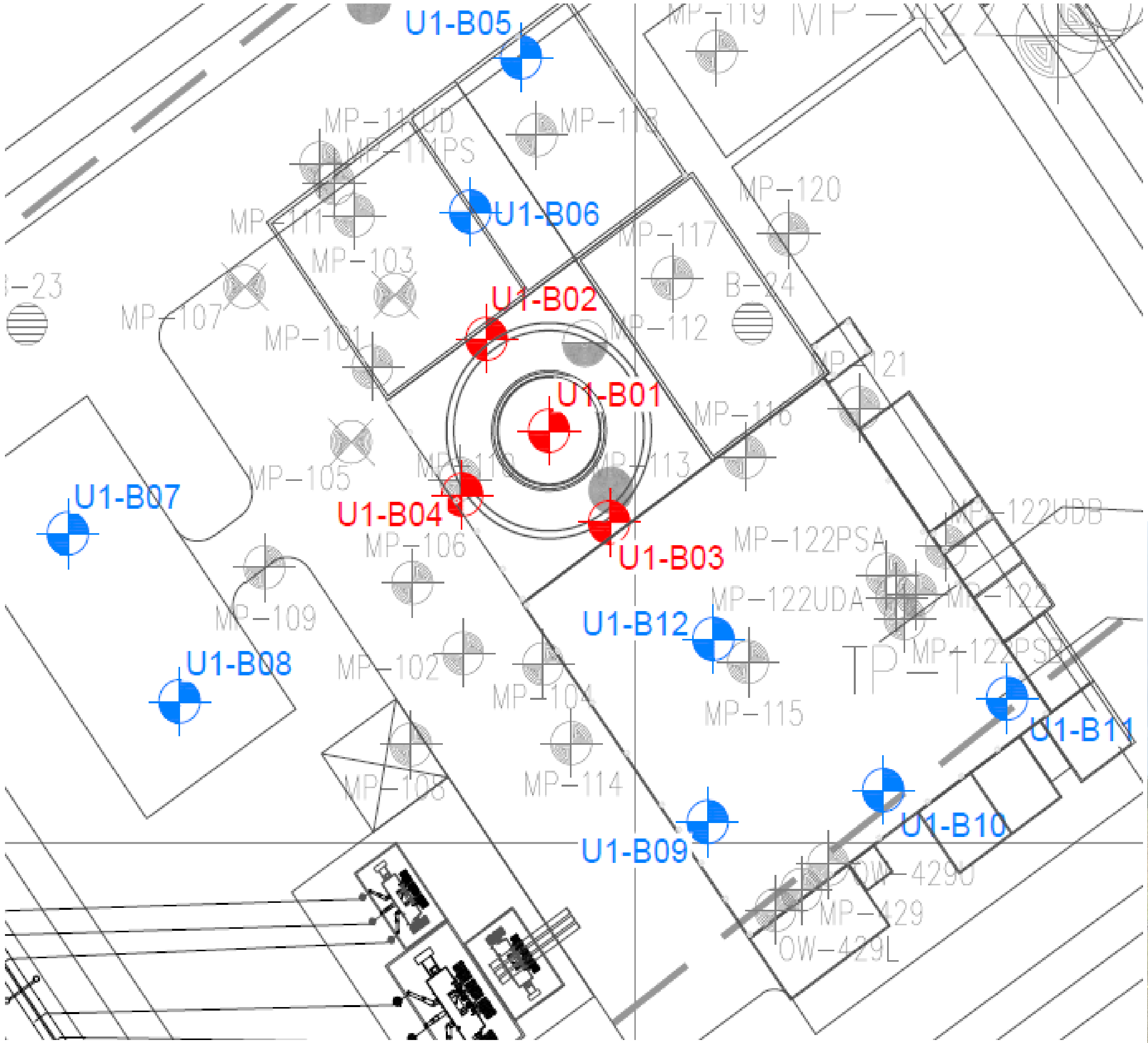
TVA TENNESSEE VALLEY AUTHORITY


Barge Unloading Facility
Grassy Creek HPA
CRN-1 Disturbance Area
Zion Patrol Road
Bear Creek Road
CRM 14
CRM 15
CRM 16
CRM 17
CRM 18
CRM 19
Bear Creek Road Site Entrance
58
Hensley Cemetery
Clinch River arm of the Watts Bar Reservoir
1,000
0
1,000
Feet
Fenced Area (17.5 acres)
Plant Infrastructure (8.6 acres)
Optional Quarry (40.2 acres)
Construction/Operational Support Area (100.7 acres)
Pond (14.8 acres)
Kingston FP-Bethel Valley #2 161-kV Transmission Line (TL)
Barge Unloading Facility
Bear Creek Road Site Entrance
Grassy Creek Bridge
CRN Site Access Road
Grassy Creek Habitat Protection Area
Bull Run FP-Watts Bar NP 500-kV Corridor
River Road
Optional Quarry
Concrete Batch Plant
Intake Area
Stormwater Basin
Fire Protection Enclosure and Tanks, Potable Water, Demineralization Water Treatment and Tanks
Administration Building
Discharge/ Diffuser Pipes
Discharge Structure
Blowdown Holding Pond
CRN-1 Power Block
Cooling Tower
Vent Stack
Wastewater Collection Pond
Potential Meteorological Tower Site
Switchyard
Clinch River arm of the Watts Bar Reservoir
Stormwater Basin
Hensley Cemetery
TVA TENNESSEE VALLEY AUTHORITY

PSAR Chapter 2 – Site Characteristics and Site Parameters

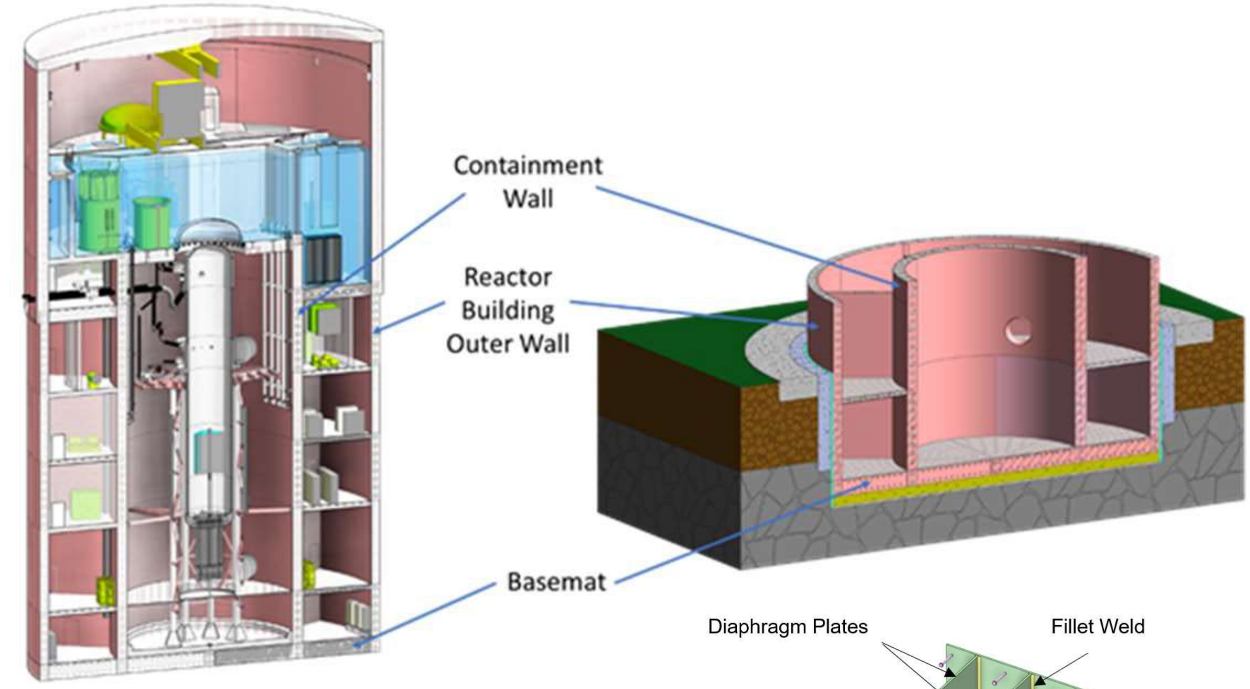
- Dispositions ESP-006 Permit Conditions and COL Action Items
- Updated CRN Site Characteristics and Parameters
- Aspects of CRN ESPA Site Safety Analysis Report Incorporated by Reference
- PSAR Table 1.8-1 provides a cross reference of Site Safety Analysis Report information that is incorporated by reference into this PSAR:
  - 2.0 Plant Parameter Envelope Evaluation
  - 2.1 Geography and Demography
  - 2.2 Nearby Industrial, Transportation and Military Facilities
  - 2.3 Meteorology
  - 2.4 Hydrologic Engineering
  - 2.5 Geology, Seismology, and Geotechnical Engineering
  - 13.3 Emergency Preparedness
  - 13.6 Physical Security

CRN-1 Site Plan Confirmatory Core Bores

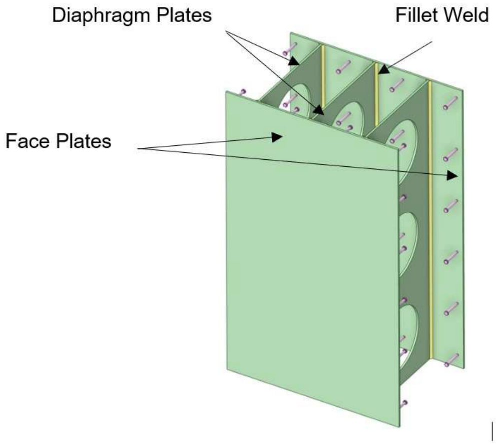


TVA TENNESSEE VALLEY AUTHORITY

Chapter 3-Design of Structures, Systems, and Components

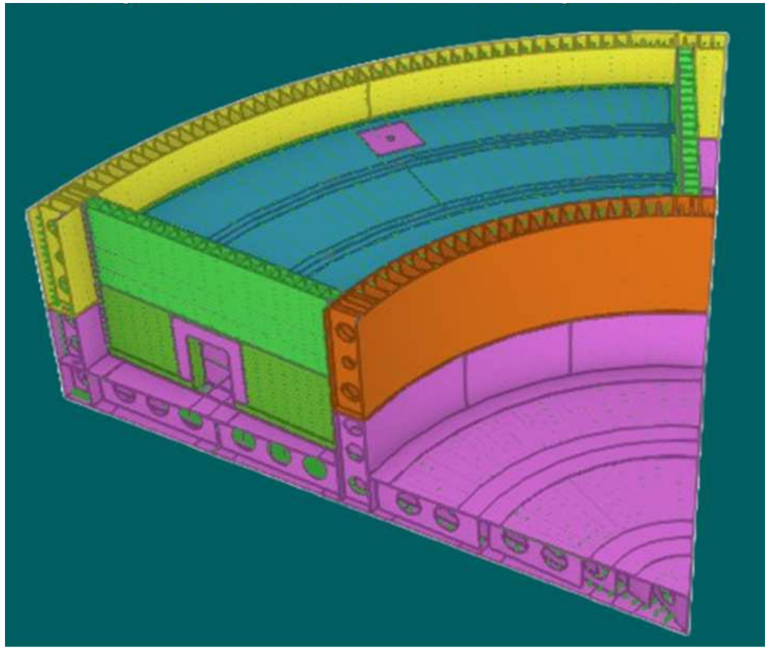


BWR-300 Reactor Building



<u>BWRX-300 Design Feature</u>

- ➢ Safety Strategy LTR in Review (Section 3.2)
- ➢ DPSC LTR Rev 3 in Review (Section 3.8)



Chapter 4 – Reactor & Chapter 5 – Reactor Coolant System and Connected Systems

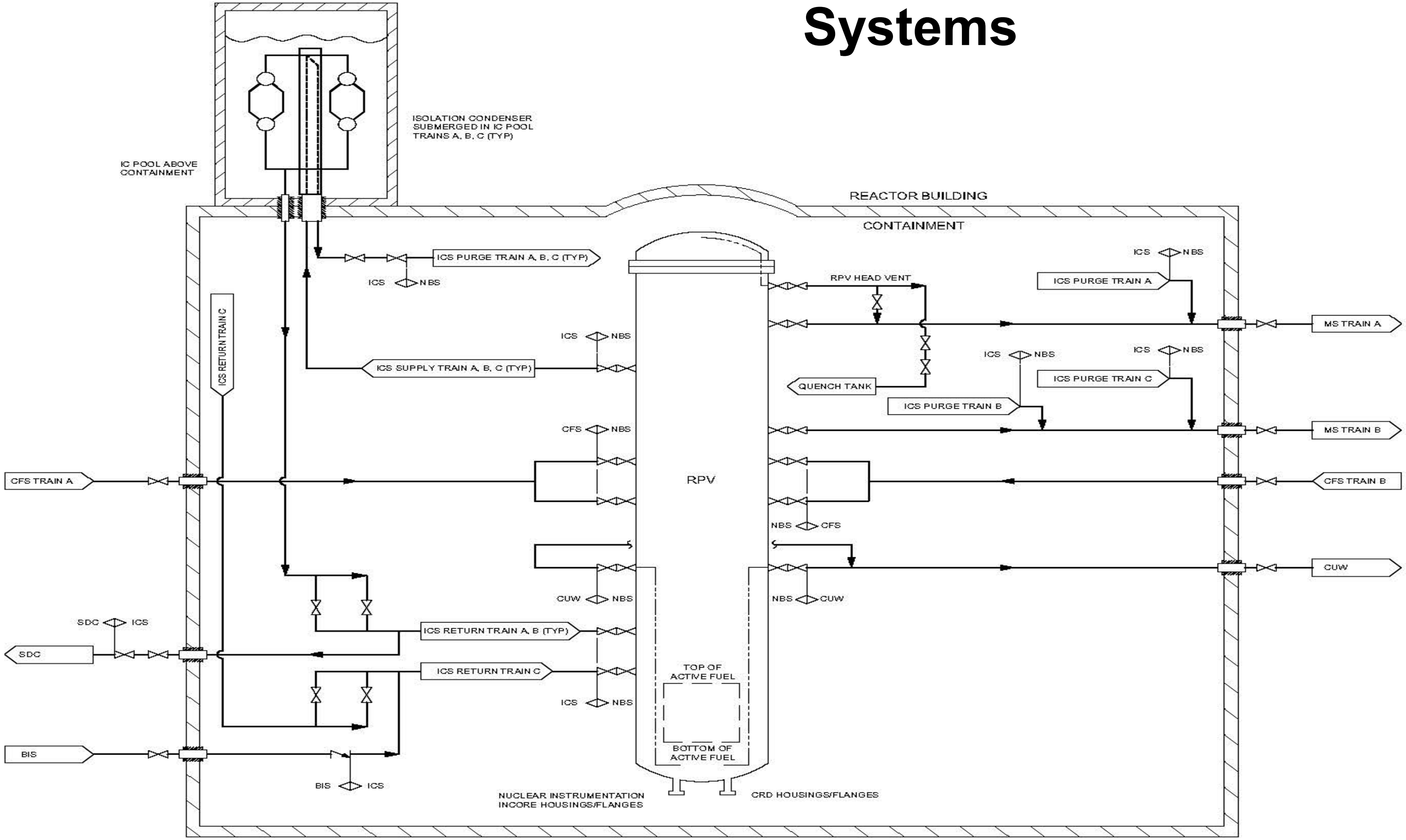


Figure 5.1-2 BWRX-300 Reactor Coolant Pressure Boundary Simplified Diagram

<u>BWRX-300 Design Feature</u>

- Natural Circulation BWR
- Increased RPV height
- Tall chimney
- Reactor Isolation Valves
- Flow Stability LTR In Review
- Reactor Isolation Valves

Chapter 6 – Engineered Safety Features

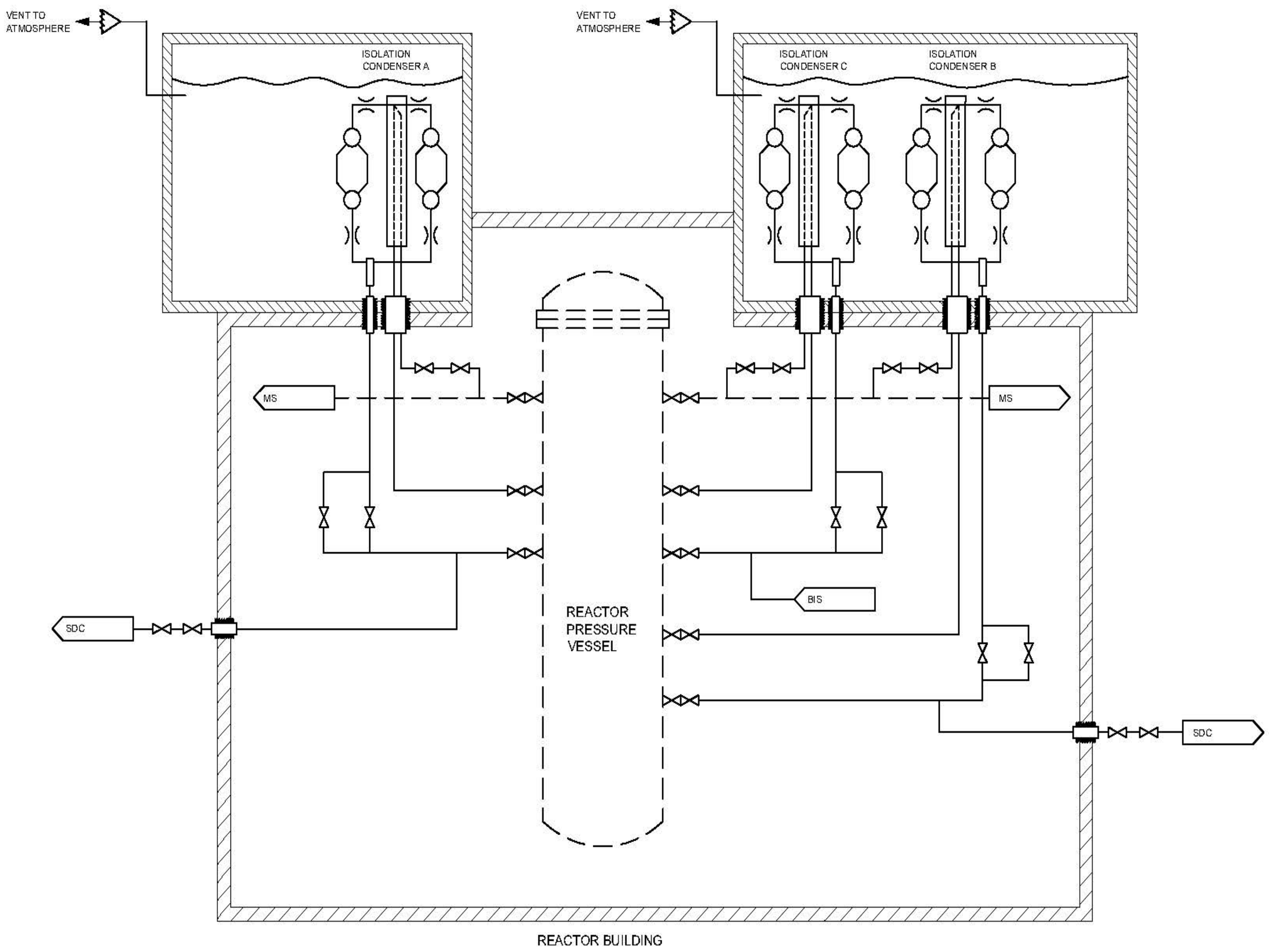
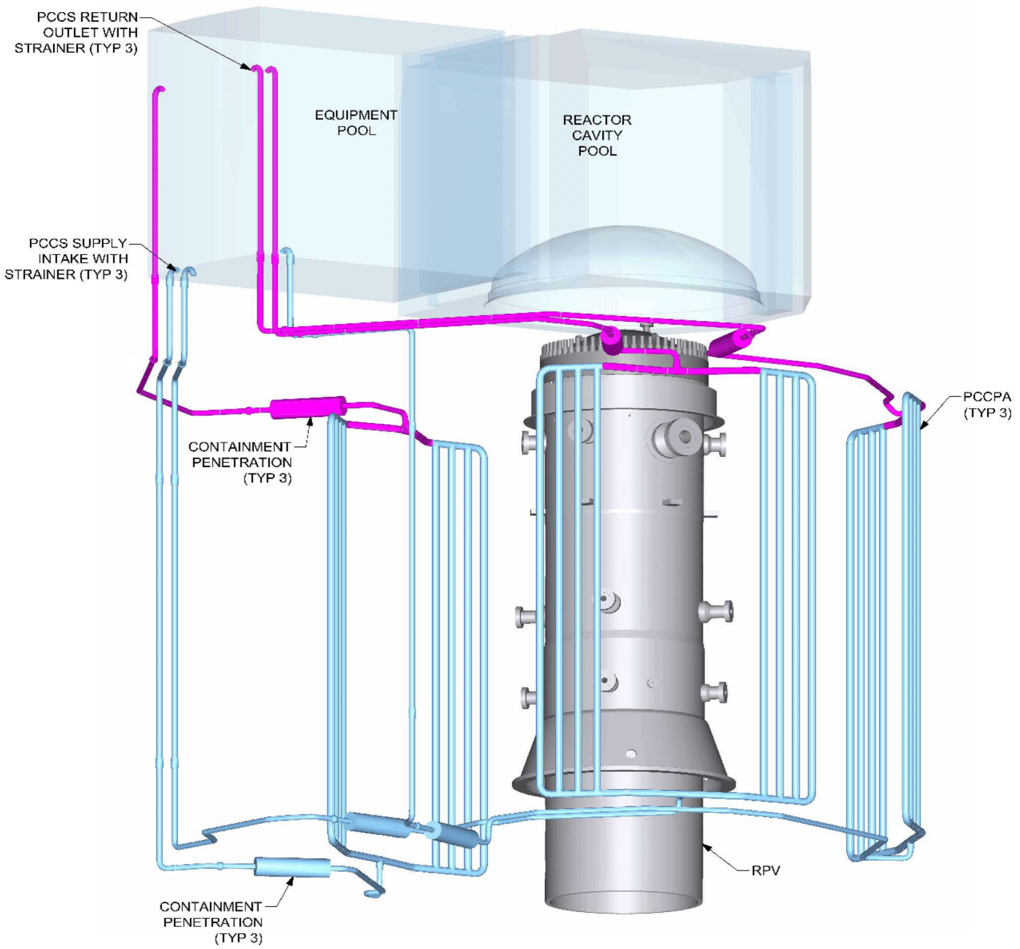


Figure 6.3-1 Isolation Condenser System Simplified Diagram

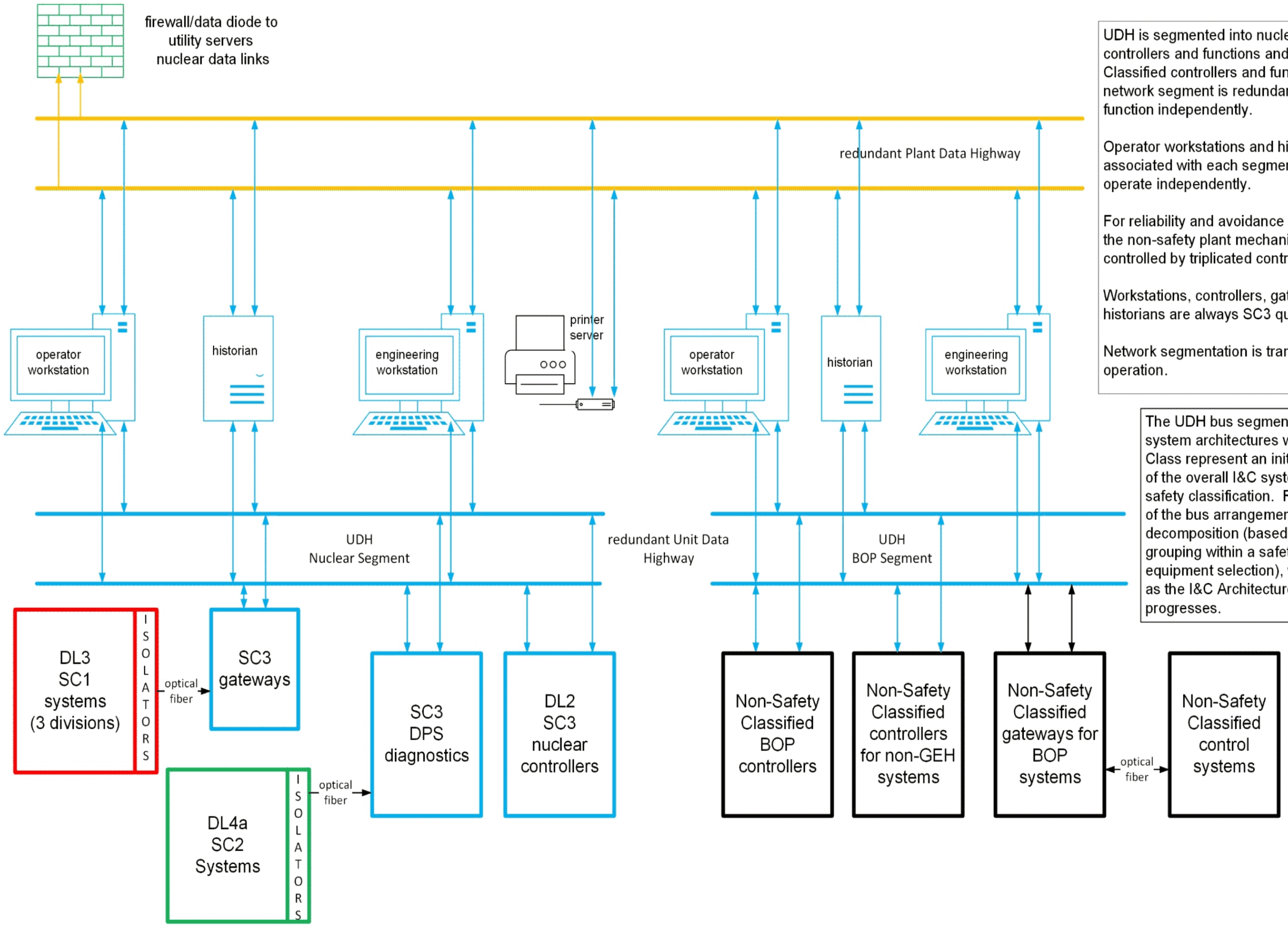


Note: Valves are not shown.

Figure 6.2-1 Passive Containment Cooling System Configuration

TVA TENNESSEE VALLEY AUTHORITY

Chapter 7 – Instrumentation and Controls



UDH is segmented into nuclear SC3 controllers and functions and BOP Non-Safety Classified controllers and functions. Each network segment is redundant and can function independently.

Operator workstations and historians are associated with each segment and also operate independently.

For reliability and avoidance of AOO failures, the non-safety plant mechanical systems are controlled by triplicated controllers.

Workstations, controllers, gateways and historians are always SC3 quality.

Network segmentation is transparent in normal operation.

The UDH bus segmentation and the system architectures within a Safety Class represent an initial decomposition of the overall I&C systems based on safety classification. Further refinement of the bus arrangement and system decomposition (based on functional grouping within a safety class and equipment selection), will be performed as the I&C Architecture design process progresses.

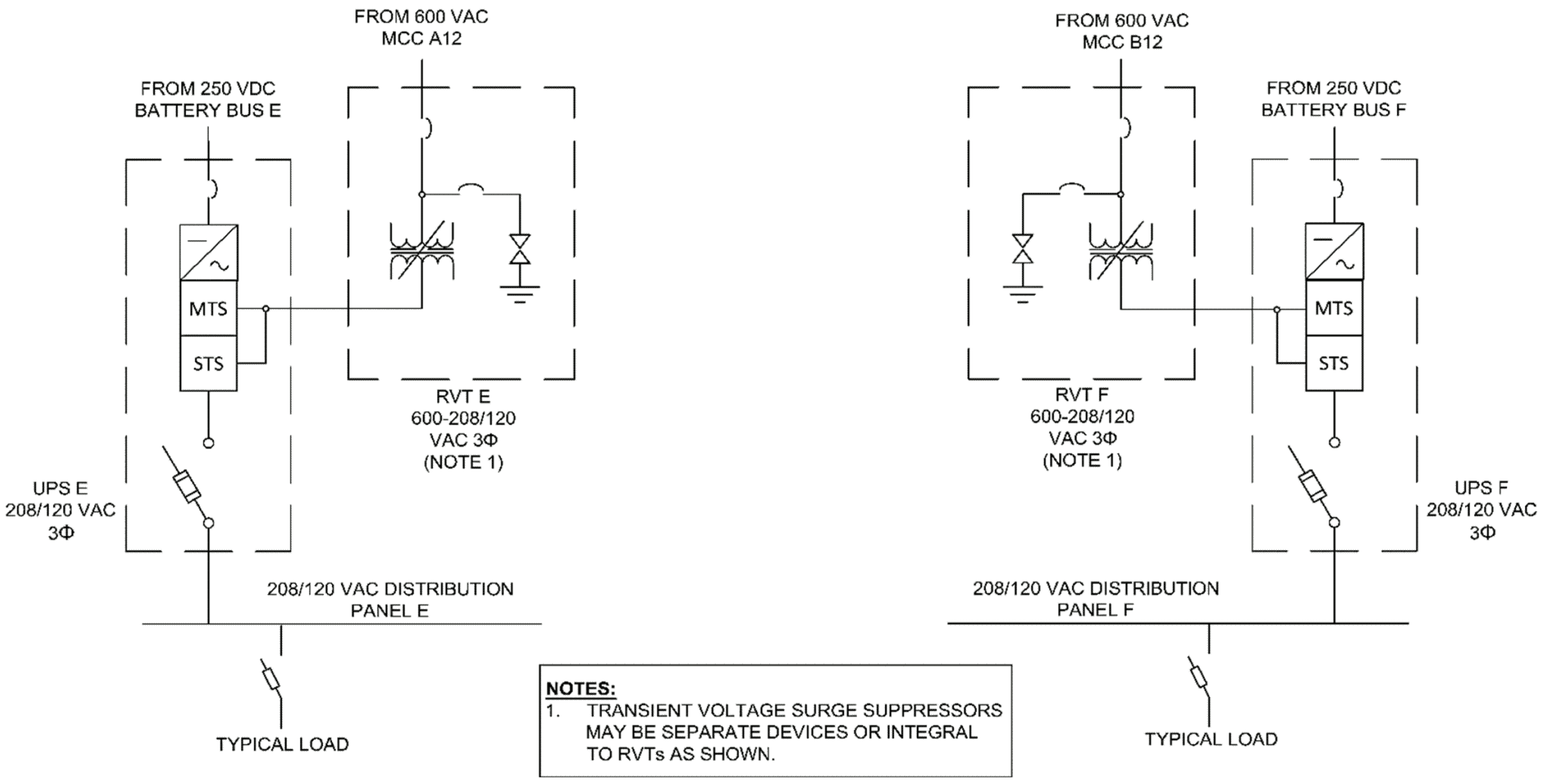
Chapter 8 – Electric Power

Chapter 8 Contents Includes:

- Offsite and Onsite Power Systems
- Uninterruptable Power Supply

BWRX-300 Design Feature

- The BWRX-300 does not require AC power to reach a safe, stable shutdown following an Anticipated Operational Occurrence or a Design Basis Accident
- Stored energy via batteries is provided:
    1. Ensure that all functions that maintain the plant in a safe condition are available
    2. Monitoring equipment can be powered for at least 72 hours following a Design Basis Accident.



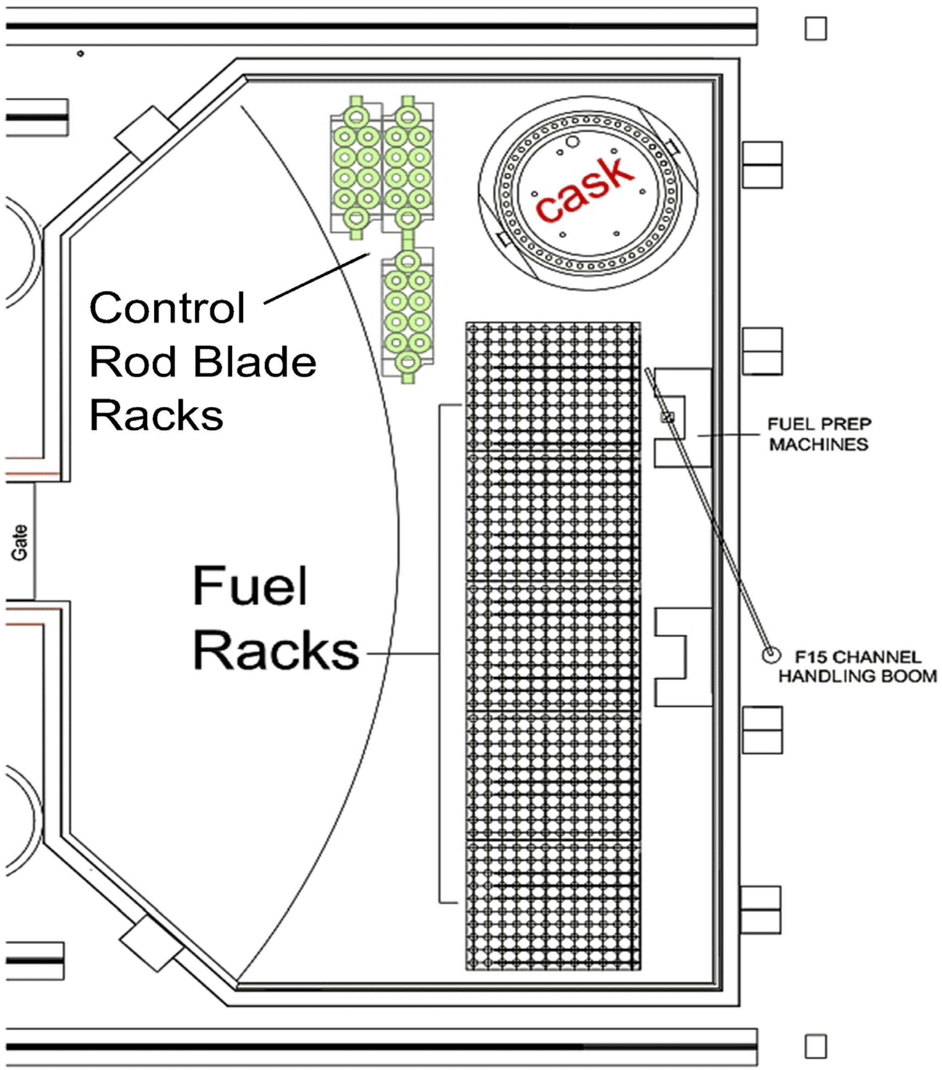


Figure 9.1-2 Fuel Pool Arrangement

Chapter 9- Auxiliary Systems

<u>BWRX-300 Design Feature</u>

- Multiple credited Ultimate Heat Sinks
- BWRX-300 water is strategically located during operations in SC1 pools to last for 7 days until FLEX/EME replenishment

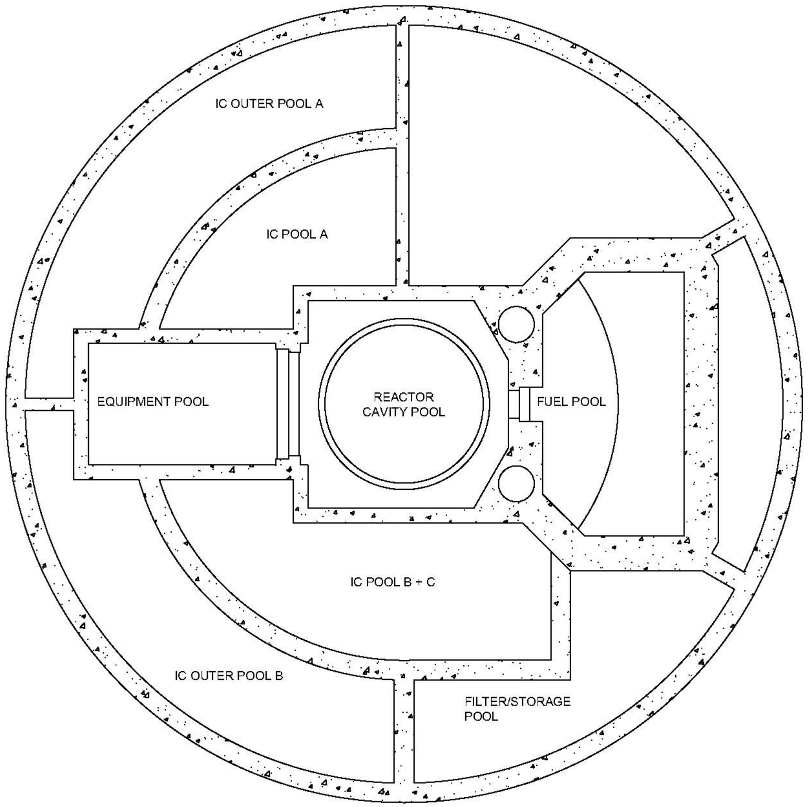


Figure 9.2-3 Ultimate Heat Sink Pools Simplified Diagram

Chapter 10 – Steam and Power Conversion System

<u>Chapter Contents Includes</u>:

- ➢ Turbine Generator
- ➢ Main Steam System
- ➢ Additional Steam and Power Conversion Systems

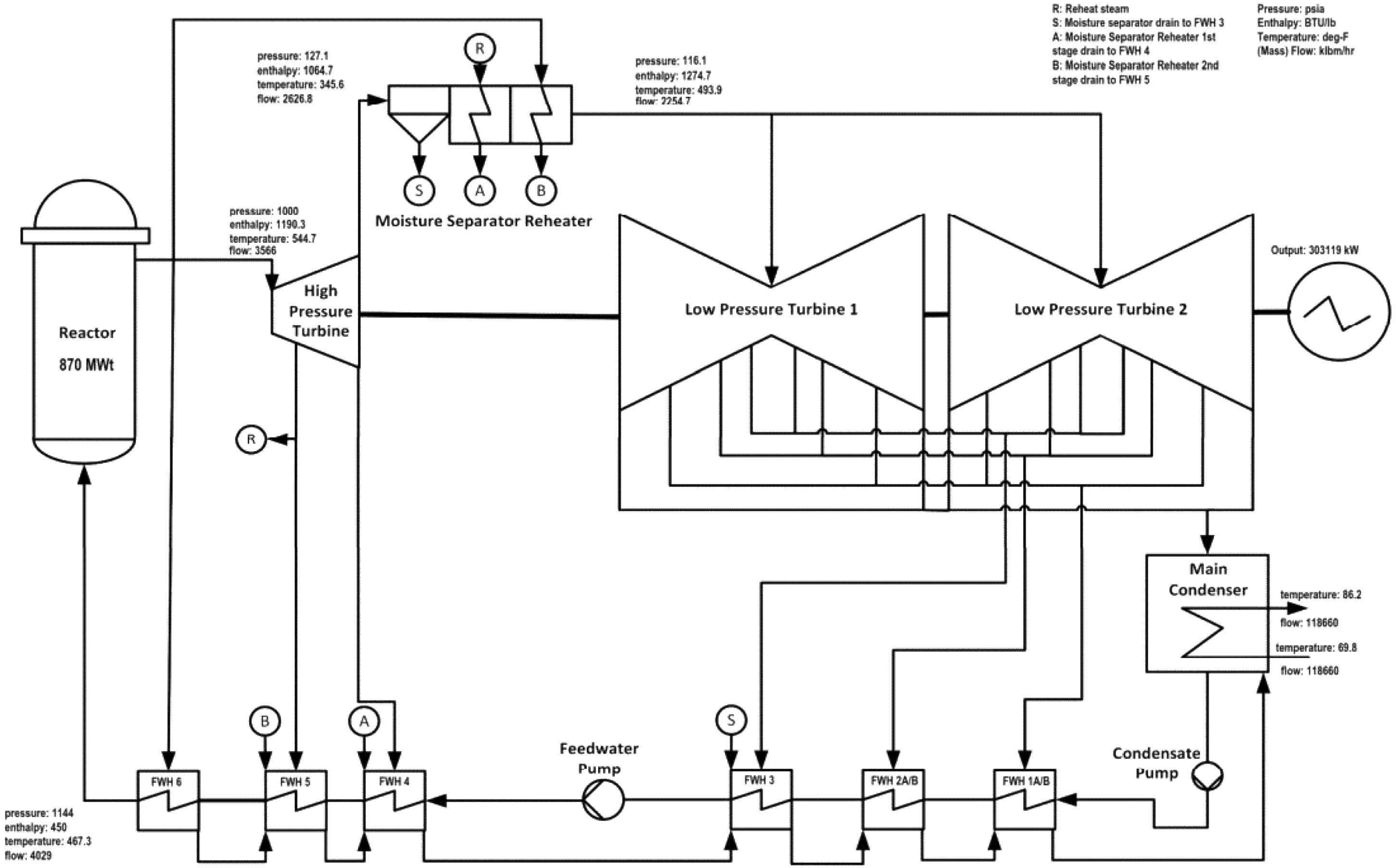


Figure 10.1-1 Simplified Flow Diagram with Representative Heat Balance of the Steam and Power Conversion System

TVA TENNESSEE VALLEY AUTHORITY

Chapter 11 – Radioactive Waste Management

Chapter Contents Includes:

- Source Terms
- Liquid Waste Management System
- Gaseous Waste Management System
- Solid Waste Management System
- Process Radiation Monitoring

Chapter 12 –Radiation Protection

Chapter Contents Includes:

- Occupational Radiation Exposure ALARA
- Radiation Sources
- Radiation Protection Design Features
- Dose Assessment
- Health Physics Program

Chapter 13 – Conduct of Operations

Chapter Contents Includes:

- Organizational Structure
- Training
- Emergency Preparedness
- Operational Programs
- Plant Procedures
- Physical Security
- Fitness for Duty

Chapter 14 – Initial Test Program

Chapter Contents Includes:

- Scope of Initial Test Program
- Design Features that are Specific, Unique or First of a Kind
- Conformance of Test Programs with Regulatory Guides
- Test Program Schedule
- Augmenting Staff During Test Program

Chapter 15 – Safety Analyses

Chapter Contents Includes:

- Considerations of the BWRX-300 Safety Analysis
- Identification, Categorization and Grouping of Postulated Initiating Events and Accident Scenarios
- Safety Objectives and Acceptance Criteria
- Human actions
- Deterministic Safety Analyses
- Probabilistic Safety Assessment
- Results of Deterministic Safety Analyses and Probabilistic Safety Assessment

BWRX-300 Design Feature

- Re-characterization of Safety Related/Non-Safety Related to the Safety Class 1, 2, 3, N structure

Chapter 16 – Technical Specifications

Chapter Contents Includes:

- Preliminary Safety Analysis Report Requirements
- Regulatory Guidance for Preliminary Technical Specification Contents
- Conformance with Industry Standards and Practices
- Methodology for Selection of Preliminary Technical Specification Contents
- Results of Selection Methodology Application

Chapter 17 – Quality Assurance

Chapter Contents Includes:

- Quality Assurance During Design and Construction Phases
- Design Reliability Assurance Program
- Quality Assurance Program Description-New Reactor Applicants

Topical Report - NNP-TR-001-NP

- Quality Assurance Program Description for TVA New Nuclear incorporated by reference.
- Final Safety Evaluation contains Limitations and Conditions (PSAIs) and are disposition in Chapter 17.5

Enclosure 4 – Exemptions and Variances

Exemptions

- Reactor Vessel Material Surveillance Program

Variances

- CRN ESP VAR 2.0-1 Site Grade Level
- CRN ESP VAR 2.0-2 Ground Water Level
- CRN ESP VAR 2.0-3 Single Unit Thermal Megawatts
- CRN ESP VAR 2.1-1 2020 Census Data
- CRN ESP VAR 2.2-1 Nearby Industrial, Transportation and Military Facilities
- CRN ESP VAR 2.4.12-1 Groundwater Level Models
- CRN ESP VAR 2.4.12-1 C-1Groundwater Vistas Version 8.19 Build 4

Questions/Comments/Actions

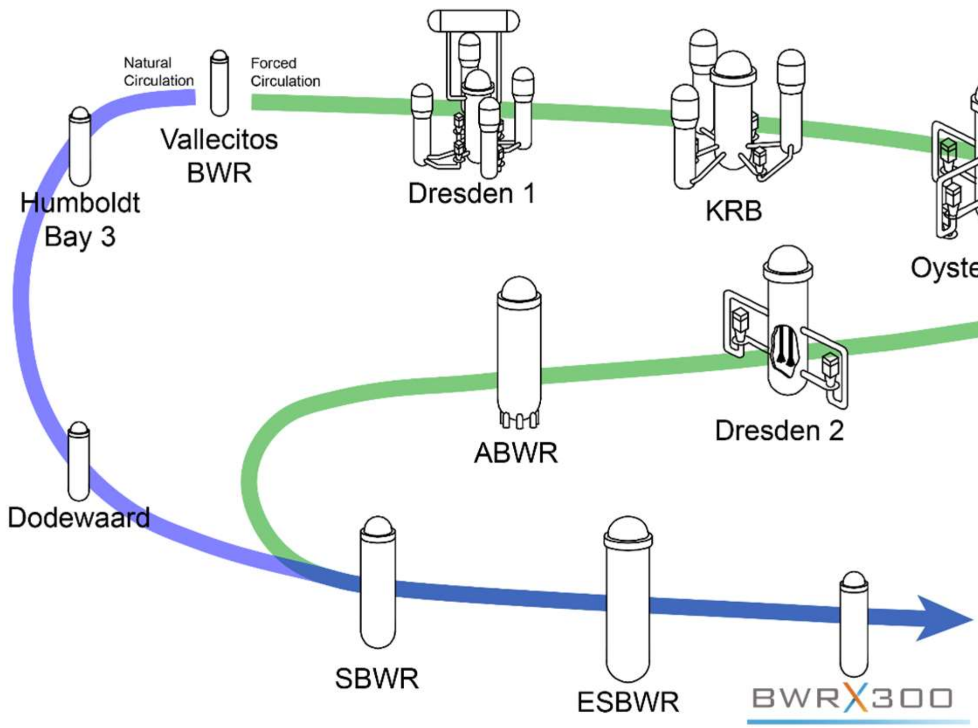
TENNESSEE
VALLEY
AUTHORITY

BWRX-300 DESIGN

Outline

- BWRX-300 Design Overview
- Unique design features for:
  - Reactor Pressure Vessel (RPV)
  - Reactor Isolation Valves (RIVs)
  - Isolation Condenser System (ICS)
  - Passive Containment Cooling System (PCCS)

Boiling Water Reactor (BWR) Innovation



- BWR concept developed in the 1950s
- Continuous evolution in the design
- Main changes related to:
  - Steam cycle
  - Recirculation flow
  - Nuclear fuel
  - Containment

BWRX-300 Design Overview

GE VERNOVA

Size

~300 MWe gross electrical output

RPV inner diameter ~ 4 meters

RPV height ~ 27 meters

240 bundles of GNF2 fuel

57 control rods

Passive Design

Safety Category 1 functions are not dependent on AC generated sources of power nor operator action to control reactivity, remove heat from the fuel, and confine radioactive material for 72 hours following a design basis accident

Select Key Features

Natural circulation BWR with increased height relative to a forced circulation BWR

RPV contains tall chimney, nozzles are well above Top of Active Fuel (TAF), and RIVs are attached directly to RPV

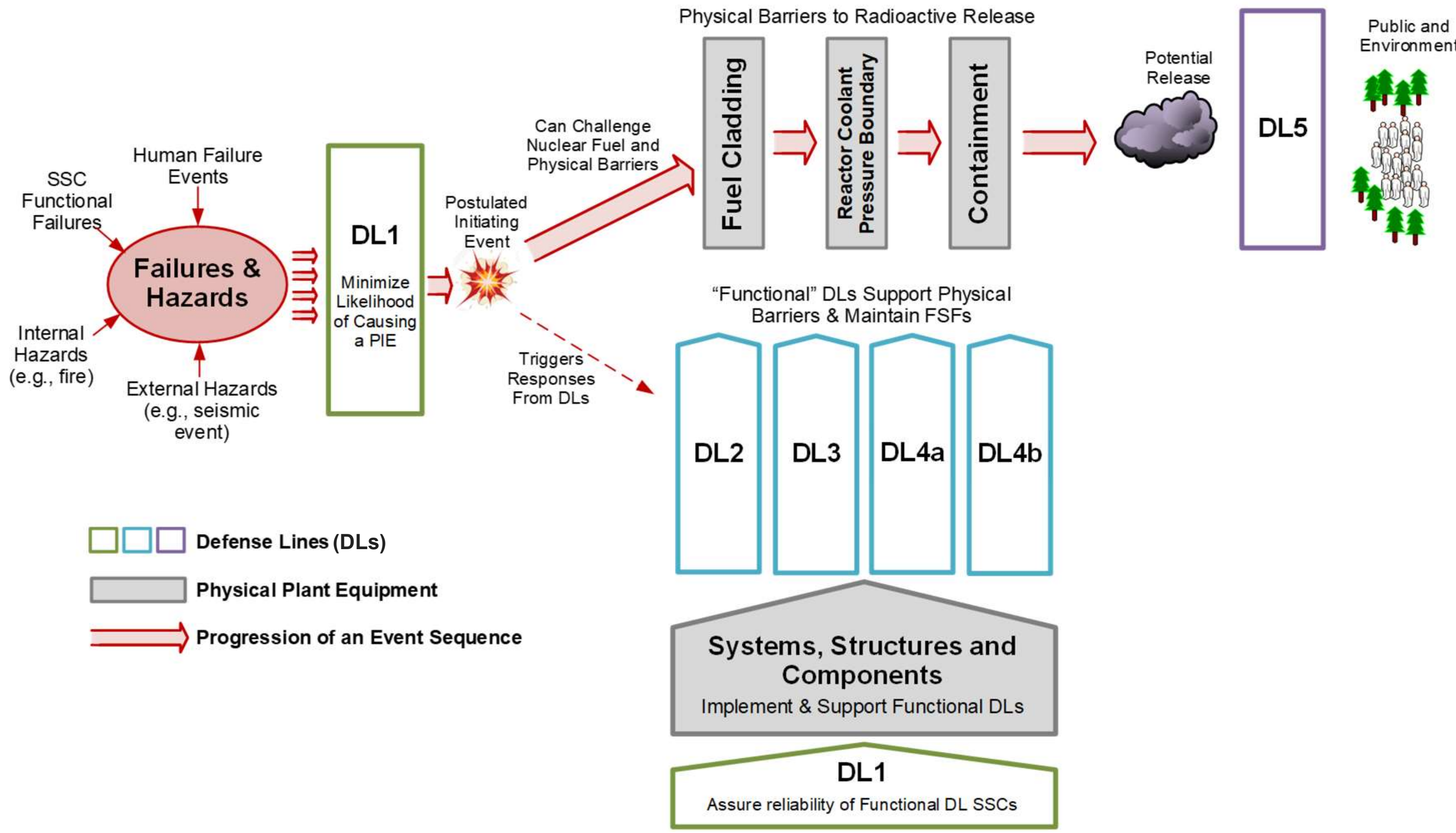
Dry, nitrogen inerted containment, which is cooled passively

Steel-Plate Composite Containment Vessel (SCCV)

Emergency Cooling System is made up of ICS and RIVs

Overpressure protection is provided via ICS and reactor scram function

Defense In Depth … Built Into The Design From The Start



NRC Approved Licensing Topical Reports (LTRs) for BWRX-300

GE VERNOVA

NEDC-33910P-A, BWRX-300 RPV Isolation and Overpressure Protection (NRC Final Safety Evaluation Report (SER) Issued 11/18/2020)

Describes design requirements, acceptance criteria, and regulatory basis for RPV isolation and overpressure protection design functions for mitigation of loss-of-coolant accidents (LOCAs) and RPV overpressure events. LTR established ECCS for BWRX-300 as ICS and RIVs, and established overpressure protection to be made up of reactor scram and ICS functions. This allowed for the elimination of an automatic depressurization system, suppression pool, additional water inventory source, relief valves, and safety valves.

NEDC-33911P-A, BWRX-300 Containment Performance (NRC Final SER Issued 3/12/2021)

Addresses physical design requirements for new dry, inerted containment design (including containment vessel, containment penetrations and PCCS), and acceptance criteria requirements (design basis pressures and temperatures) for containment performance following the specified design basis accidents.

NEDC-33912P-A, BWRX-300 Reactivity Control (NRC Final SER Issued 1/12/2021)

Describes design requirements, acceptance criteria, and regulatory basis for reactivity control functions for shutting down the reactor following anticipated operational occurrences and design basis accidents. Allows removal of safety-related standby liquid control system from design, as one is not needed to comply with NRC Anticipated Transient Without Scram (ATWS) regulations.

NRC Approved LTRs for BWRX-300

NEDC-33922P-A, BWRX-300 Containment Evaluation Method (NRC Final SER Issued 4/27/2022)

Addresses development of and qualification of analytical methods for determining containment response (calculated containment pressures and temperatures over time) after a design basis accident for comparison with acceptance criteria of NEDC-33911P-A.

NEDO-33914-A, BWRX-300 Advanced Civil Construction and Design Approach (NRC Final SER Issued 4/27/2022)

Describes regulatory basis, analytical methods, design and inspection requirements, acceptance criteria and guidelines specific to the innovative approaches implemented for design and construction of the BWRX-300 Reactor Building vertical shaft design.

Applicable Limitations and Conditions (L&Cs) from previously approved LTRs are addressed in TVA PSAR

BWRX-300 LTRs Currently Under NRC Review

NEDC-33926P, BWRX-300 Steel-Plate Composite Containment Vessel (SCCV) and Reactor Building (RB) Structural Design (Initially Submitted to NRC 5/4/2023)

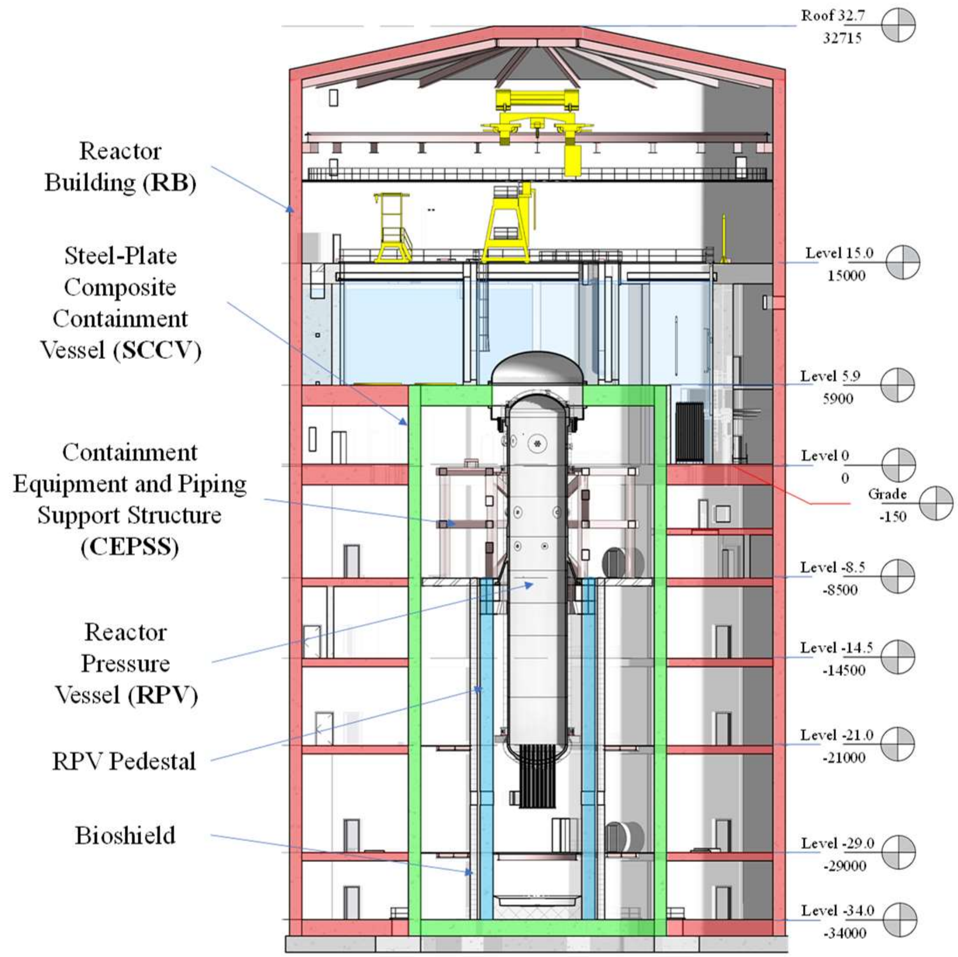
Seeks NRC approval for

(1) The design approach and methodology of Diaphragm Plate Steel-Plate Composite (DP-SC) structural elements for the Seismic Category I SCCV and RB structures,

(2) Requirements for the material, fabrication, construction, inspection, examination and testing of the DP-SC modules for the SCCV and RB structures,

(3) Proposed criteria and requirements for materials, design, fabrication, construction, inspection, examination, and testing for the SCCV adapted from specific Section III requirements, and

(4) Modified criteria and requirements for material, design, analysis, fabrication, construction, inspection, examination, and testing of non-containment Seismic Category I structural members, including slabs and curved walls, built using DP-SC modules



BWRX-300 LTRs Currently Under NRC Review

NEDC-33934P, BWRX-300 Safety Strategy (Initially Submitted to NRC 3/8/2024)

The BWRX-300 Safety Strategy applies a Defense-in-Depth design approach to achieve an internationally deployable design with an inherent high level of safety. NEDC-33934P describes the use of DL functions to mitigate design basis and beyond design basis events, and the resulting Structures, Systems, and Components (SSC) classification and seismic categorization.

NEDC-33934P, Rev. 1, seeks the following NRC approvals:

(1) BWRX-300 Safety Class 1 (SC1) SSCs are equivalent to the “safety-related SSCs” definition in 10 CFR 50.2

(2) The LTR identifies the correct set of SSCs that are applicable to GDCs involving “important to safety” or “protection system”

(3) Safety Strategy event categorization process is acceptable

(4) The LTR identifies the correct set of SSCs that are applicable to Technical Specifications Limiting Conditions for Operation Criteria

(5) Identification of Regulatory Treatment of Non-Safety Systems (RTNSS) SSCs is not necessary, as the Safety Strategy already classifies such SSCs as Safety Class 3 (SC3) or higher.

BWRX-300 LTRs Currently Under NRC Review

NEDC-34270P, BWRX-300 Stability Analysis (Initially Submitted to NRC 3/31/2025)

The BWRX-300 Stability Analysis LTR supports an applicant fulfilling L&C 5.3 from NEDC-33912P-A, *BWRX-300 Reactivity Control*, thereby conforming to NUREG-0800 Standard Review Plan (SRP) 15.9, *Boiling Water Reactor Stability*, and demonstrating compliance to the acceptance criteria provided therein. NEDC-34270P requests NRC approval of the BWRX-300 stability analysis, which utilizes implicit numerical integration for channel components and a nominal core wide Decay Ratio acceptance criterion of ≤ 0.80.

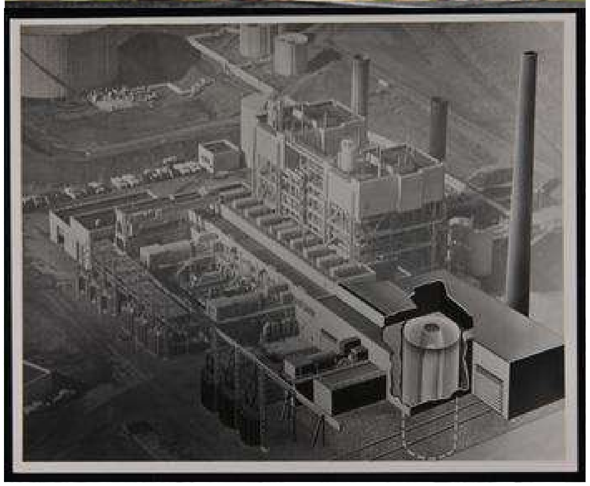
Design concepts for LTRs currently under NRC review are not expanded in this presentation, since these LTRs will get their own ACRS meeting, if required.

Unique Design Features

- Because GVH has utilized LTRs extensively for new or unique design features, ACRS has previously reviewed associated BWRX-300 design phenomena
- However, it’s been several years since ACRS has seen some of these LTRs
- Presentation will focus on the following design features:
  - RPV
  - RIVs
  - ICS
  - PCCS

Natural Circulation – Background & Overview

- Proven effective operating power reactor technology
  - EBWR (20→100 MWt), Chicago
  - Dodewaard reactor (163 MWt), Netherlands
  - Humbolt Bay 3 (215 MWt), California
- Operating BWR data gathered from Stability tests under Natural Circulation and from Recirc Pump trip events benchmarks flow at higher power (> 1000 MWt)
- Chimney two phase flow testing conducted
- Startup characteristics testing performed
- TRACG code qualification includes above data – predicts natural circulation flow well at power when flows are much higher and at decay heat powers when flows are very low.
- Core power density/size and RPV configuration to support natural circulation flow are designed to ensure thermohydraulic stability



BWRX-300 Reactor Pressure Vessel (RPV) and Internals

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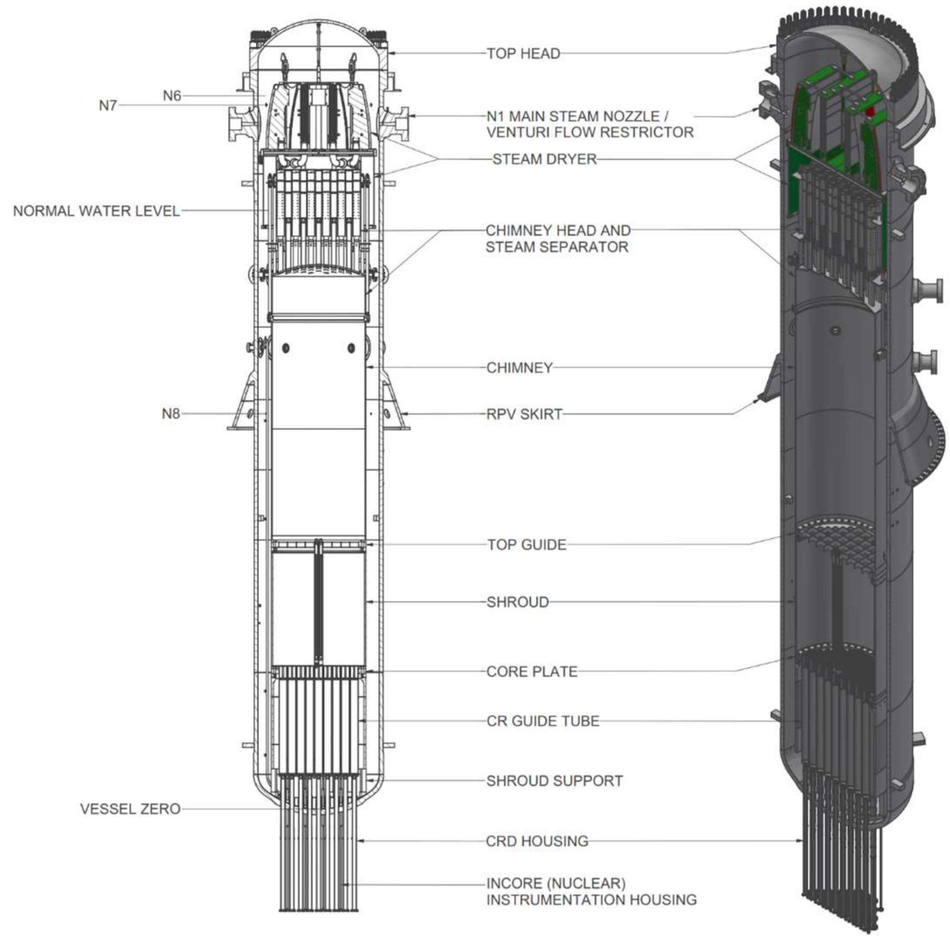
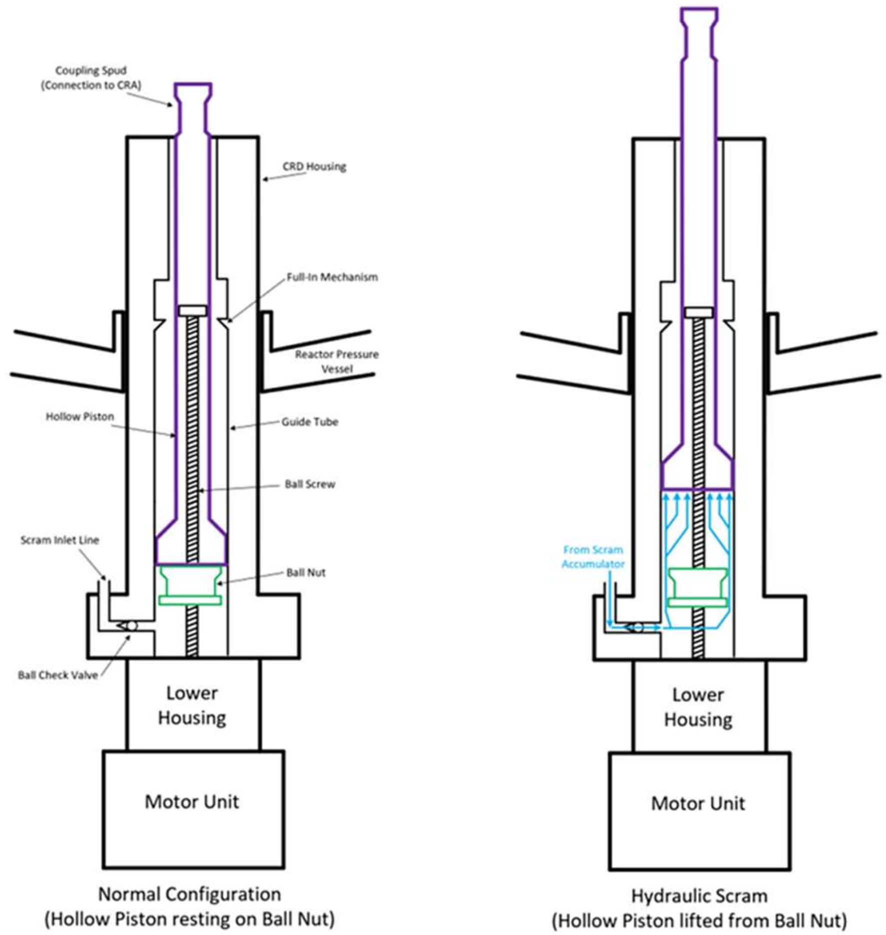
Proven Components with Operational Experience

- **RPV** same material and fabrication processes as ABWR and much of the operating BWR fleet
- **RPV** diameter and fuel assembly arrangement similar to Kernkraftwerk Mühleberg (KKM)
- **Partitionless Chimney** drives core flow.
- **Steam Dryer** has same features as ABWR and replacement dryers in operating BWRs
- **Steam Separator** is same as in BWR/6s and ABWR
- **GNF2 Fuel** is widely used
- **Control Rods** essentially same technology used in operating BWRs
- **Fine Motion Control Rod Drives (FMCRDs)** essentially same as ABWR



BWRX-300 Fine Motion Control Rod Drives (FMCRD)

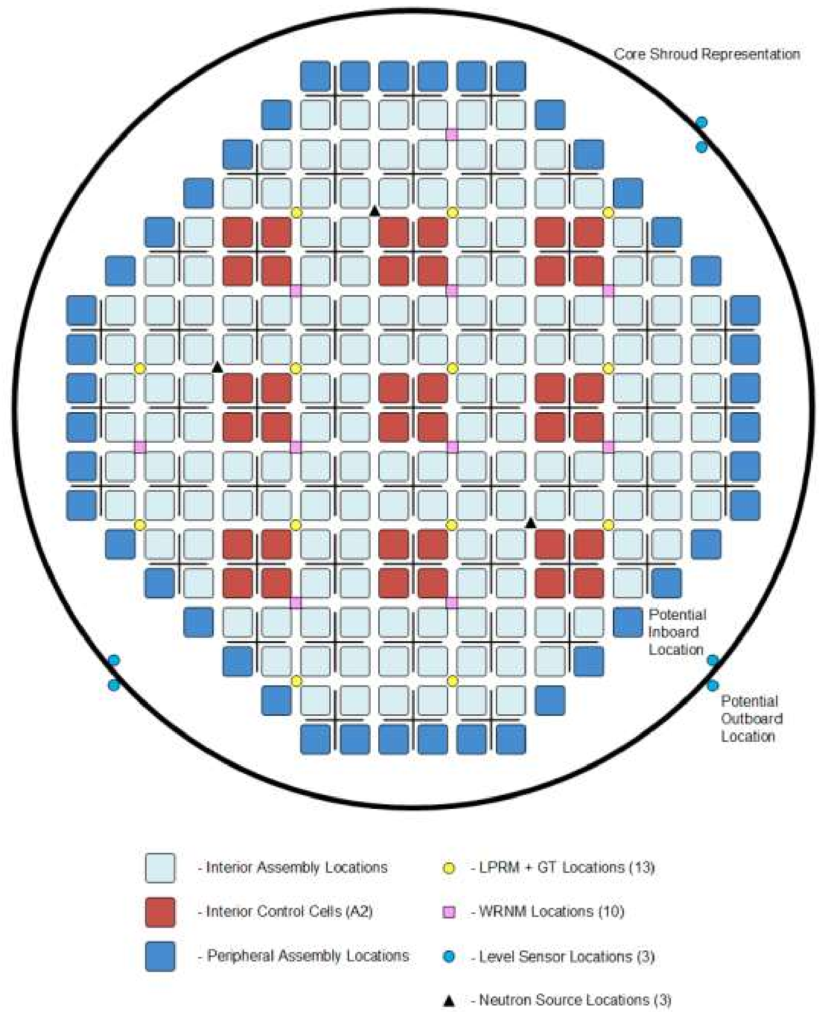
- Positive insertion means of controlling reactivity include:
    - Hydraulic scram control rod insertion function using the hydraulic control units and control rods
    - Motor-driven control rod run-in insertion function using the FMCRDs and control rods



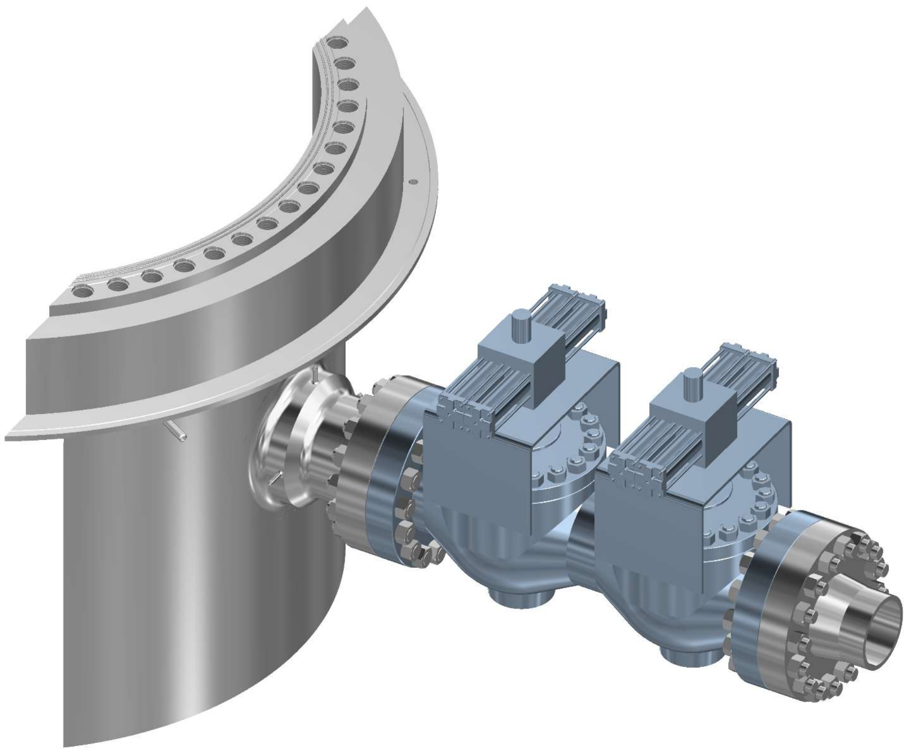
BWRX-300 Simplified View of FMCRD with Hydraulic Scram

Reactor Core Monitoring Instrumentation

- Local Power Range Monitors (LPRMs) and Wide Range Neutron Monitors (WRNMs) are distributed across the core to measure neutron flux
    - Each LPRM detector provide neutron monitoring sensitivity from ~10% core thermal power to greater than 100% reactor thermal power
    - Each WRNM detector is sensitive to neutrons from below criticality to power operation
- Fixed, in-core Gamma Thermometers (GTs) convert local gamma flux to an electrical signal, providing a diverse means of detecting core thermal power
    - GTs are used for neutron instrument calibration
    - Fixed, in-core GTs were also used in ESBWR



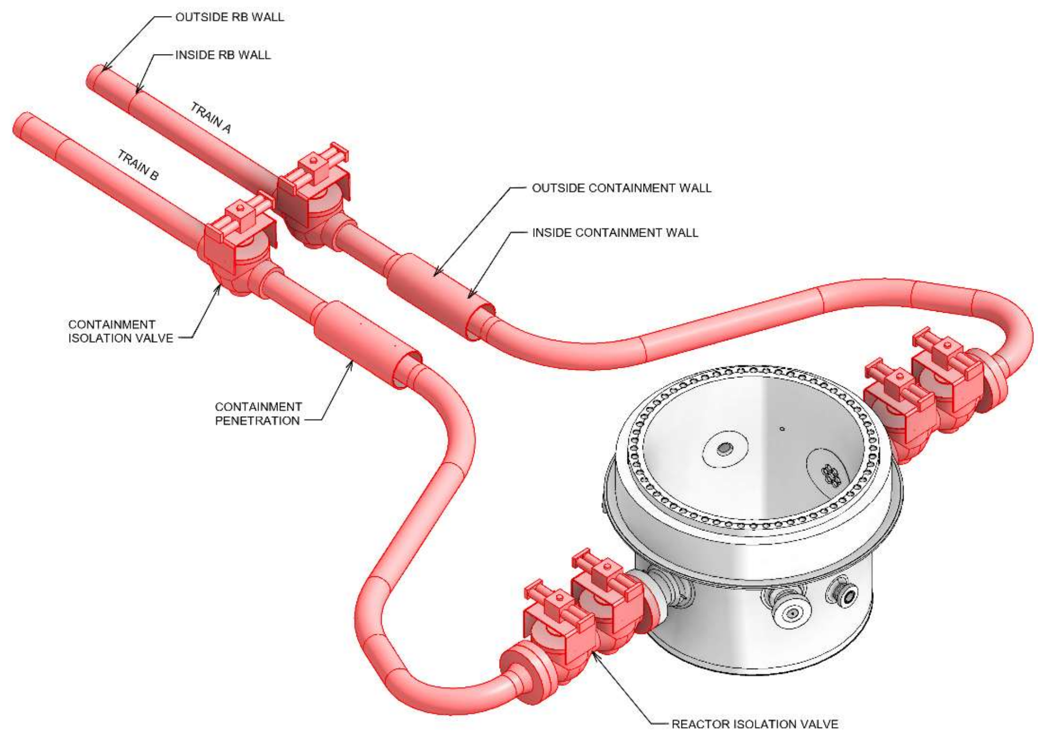
Reactor Isolation Valves (RIVs)



- All large RPV penetrations have two integral RIVs (excludes instrumentation lines)
  - Valves are installed directly on the RPV nozzles via flanged connections
  - Design consists of two valves in a single body
- RIVs are part of the reactor coolant pressure boundary and are ASME Class 1 components

Reactor Isolation Valves (RIVs)

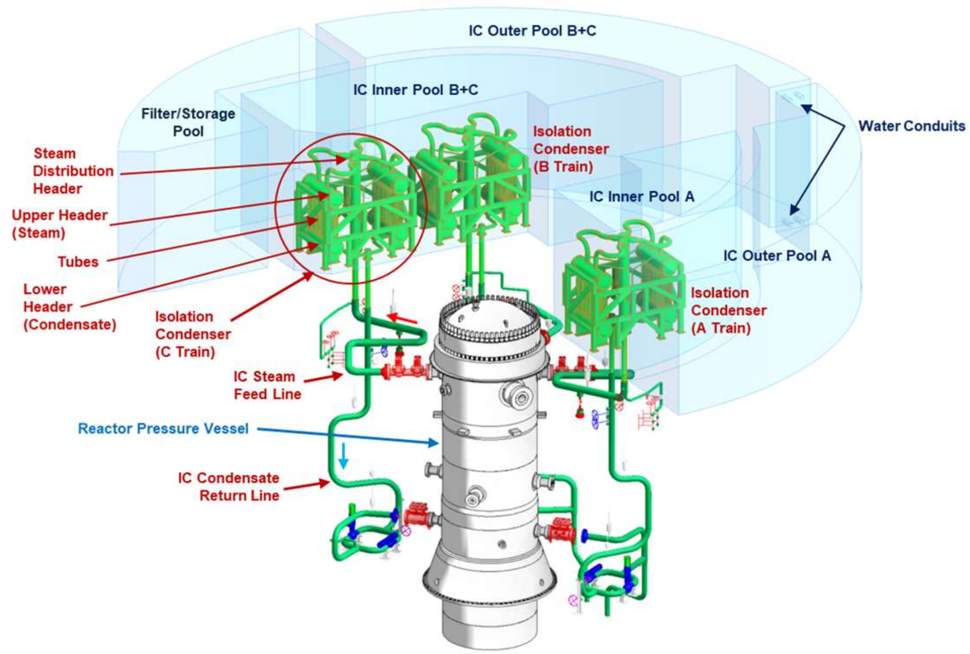
- RIVs effectively mitigate large pipe breaks
  - Coolant loss is limited by one of two RIV closure for large breaks
- RIVs are also part of the containment isolation function (i.e., are the containment isolation valve inside containment)



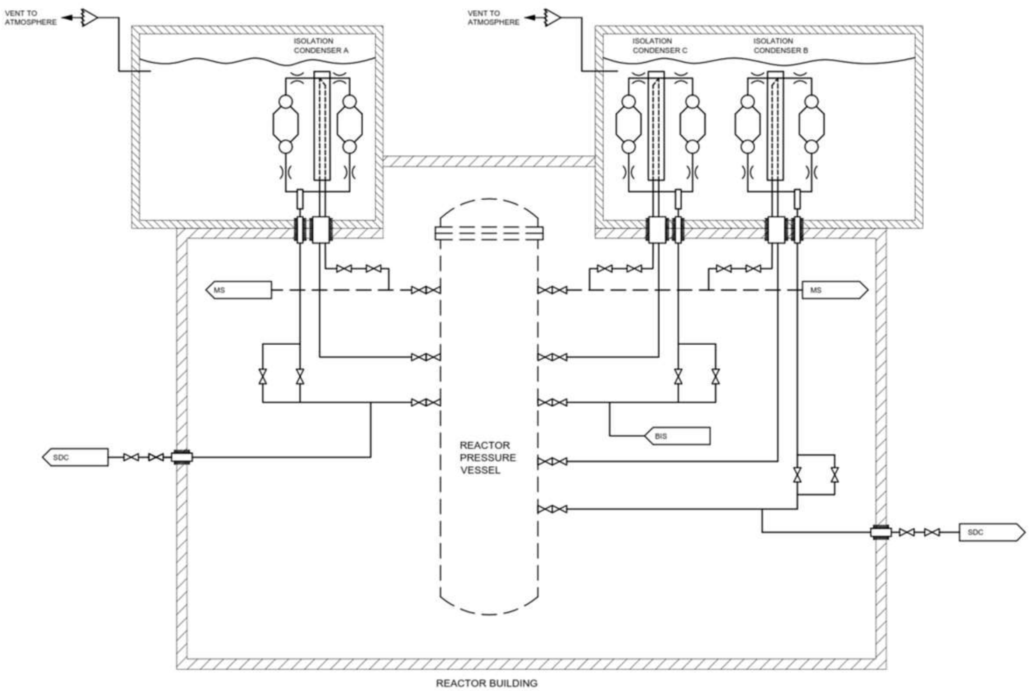
Isolation Condenser System (ICS)

- Isolation Condenser System (three trains) provides heat removal/pressure control
- Mild transient response due to large steam volume in RPV
- No need for safety relief valves – ICS along with scram function provides overpressure protection
- Only one Isolation Condenser (IC) train required to respond to the transient.
- Seven-day coping time for station blackout and with passive system response to transients and design basis accidents
- Simple actions of adding water using installed systems or FLEX after seven days to increase time indefinitely

SEVEN DAYS COPING TIME

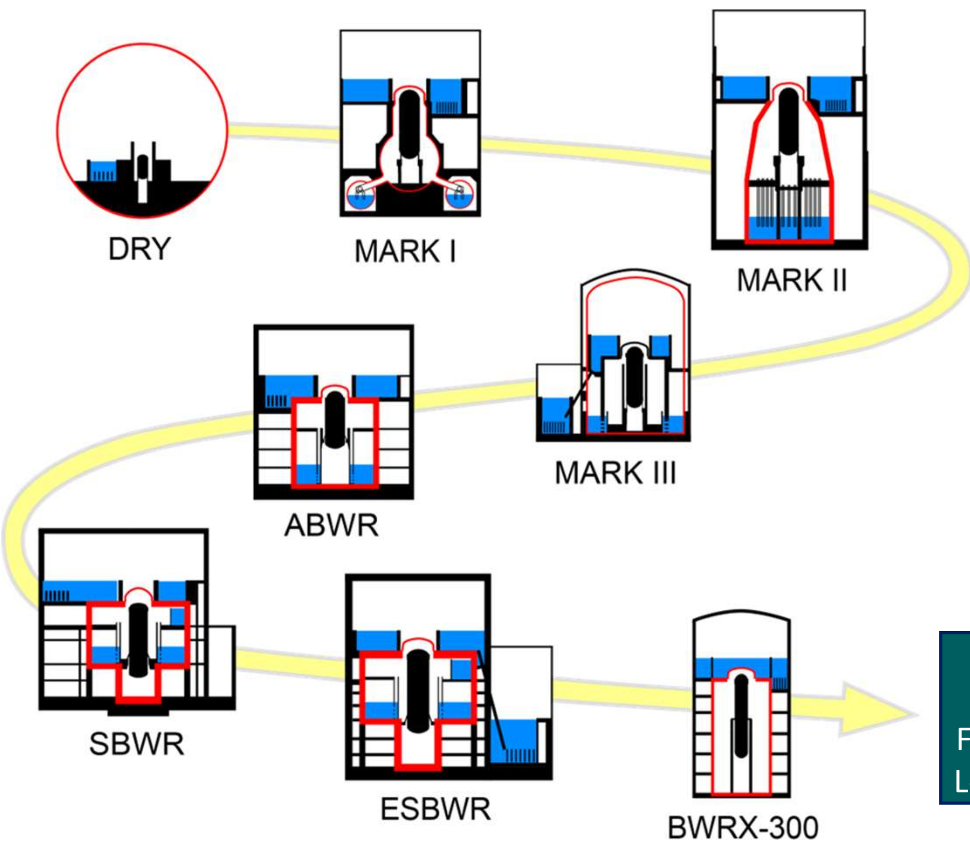


Isolation Condenser System (ICS) Functions



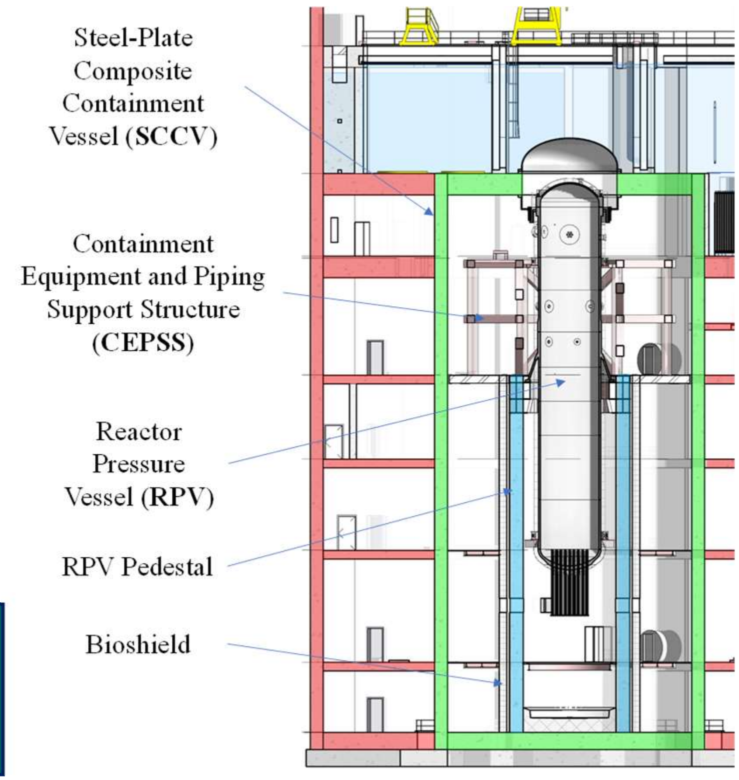
- ICS along with RIVs perform ECCS function since inventory is being retained and decay heat is being removed
- ICS in conjunction with reactor scram provides reactor pressure boundary overpressure protection when system is isolated
- ICS provides isolation capability to maintain Primary Containment integrity
- ICS returns condensate to the chimney in the RPV
- ICS provides heat removal in all modes when the RPV head is in place
- ICS mitigates pressurization transients, provides decay heat removal for isolation events, and provides pressure reduction in LOCA events to limit coolant loss

BWR Containment Design Evolution

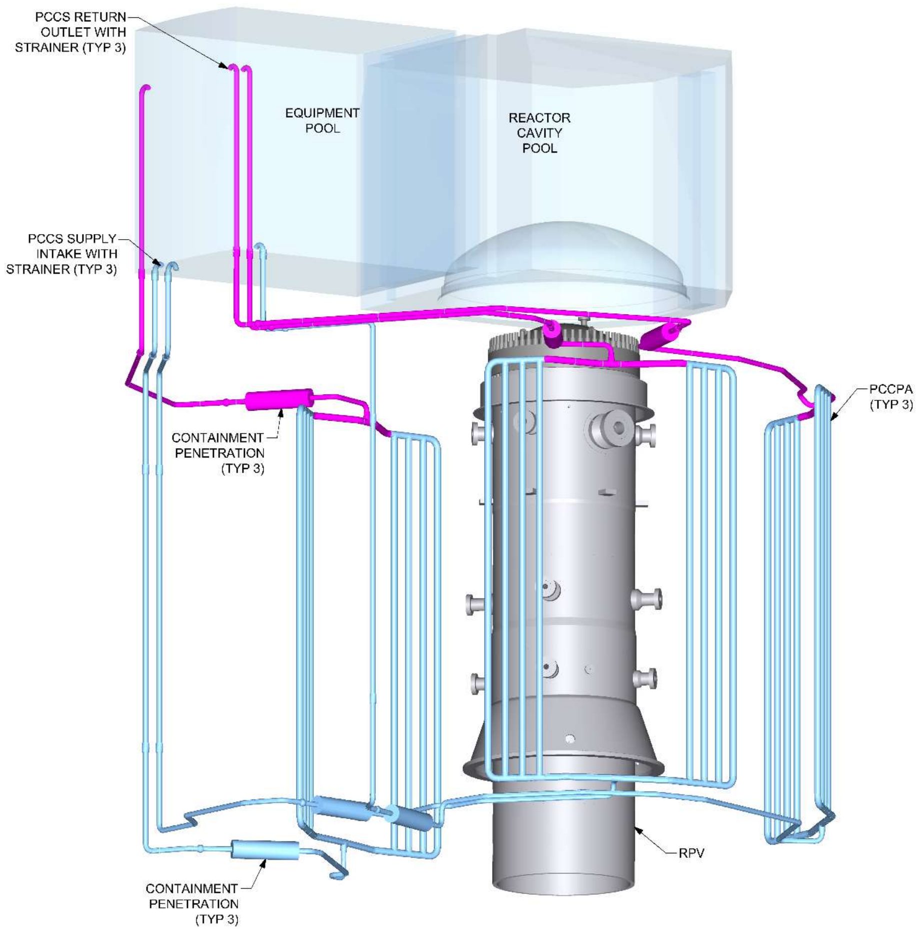


Simpler
Smaller
Faster to Build
Less Expensive

BWRX-300 has a dry containment like the earliest BWRs



Passive Containment Cooling System (PCCS)



- During normal operation heat is removed from the containment by active cooling
- Following an accident, PCCS provides containment heat removal using passive natural circulation flow
- Heat is also removed from Containment naturally through the containment head
- PCCS is always in service unless portions are manually isolated (i.e., no active components or actuation signals required to initiate or maintain function)
- Equipment pool provides the cooling source for the PCCS heat exchangers
- Three independent trains each with a Passive Containment Cooling Pipe Array (PCCPA)
- Mounted to interior of Primary Containment wall
- Piping to and from the Equipment Pool

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1. Summary

Meeting title: Clinch River CPA - Overview
Attended participants: 98
Start time: 8/20/25, 7:26:52 AM
End time: 8/20/25, 1:35:51 PM
Meeting duration: 6h 8m 59s

Average attendance time: 2h 20m 33s

2. Participants
Name
Quynh Nguyen
Allen Fetter
Ricky Vivanco
Stacy Joseph
Thomas Dashiell
William Roggenbrodt
Ravi Penmetsa
Elias Haddad
India Banks
Shandeth Walton
Thomas Scarbrough
Steven Bloom
John Honcharik
Derek Widmayer
Theresa Buchanan
John Parillo
Angie Buford
Joshua Miller
Alexandra Terres
Walt Kirchner
Stewart Bailey
Allegra Chilstrom
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Syed Haider
George Thomas
Tammy Skov
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Montague, Kelvin Jevon
Moorrees, Michele Yvette
Vesna Dimitrijevic
Dominik Muszynski (Unverified)
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Gordon Curran
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Spencer Toohill (Unverified)
Casey Emler
Jason Thompson
Fanta Sacko
Robert Martin
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dennis bley (Unverified)
Janet Riner
Andrea Torres
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Harrison Ngo
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Steven Pope
Jonathan DeJesus
Spencer Toohill (Unverified)
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Christina Antonescu
Hosung Ahn
Christopher Brown
Joseph Staudenmeier
Raul Hernandez
Karen Sida
Mary H Miller (Services - 6)
Jordan Glisan
Michael Snodderly
Wendell Morton
Sandra Walker
Matthew Yoder

Tuccillo, Karen [DEP]
Roberto Torres Davis
Hossein Nourbakhsh
Edward Stutzcage
Dennis Bley
Cory Padilla
Spencer Toohill (Unverified)
Weidong Wang
Yoshinori TAKECHI_NRA Japan (Unverified)
Getachew Tesfaye
Madeleine Arel
John Bozga

3. In-Meeting Activities
Name
Quynh Nguyen
Allen Fetter
Ricky Vivanco
Stacy Joseph
Thomas Dashiell
William Roggenbrodt
William Roggenbrodt
Ravi Penmetsa
Ravi Penmetsa
Elias Haddad
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Scott Hunnewll	TVA
Brian McDermott	TVA

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