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Oral presentation

Written submission from Canadian Coalition for Nuclear Responsibility & Prolet Inc.

In the Matter of the

Ontario Power Generation Inc.

Application for a licence to construct one BWRX-300 reactor at the Darlington New Nuclear Project Site (DNNP)

Commission Public Hearing
Part-2

January 8, 2024

Exposé oral

Mémoire du Regroupement pour la surveillance du nucléaire et de Prolet Inc.

À l'égard d'

Ontario Power Generation Inc.

Demande visant à construire 1 réacteur BWRX-300 sur le site du projet de nouvelle centrale nucléaire de Darlington (PNCND)

Audience publique de la Commission
Partie-2

8 janvier 2024

Intervenor Submission

regarding

OPG application for a license for Construction of a BWRX-300 next to an operating Darlington NGS.

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to

Canadian Nuclear Safety Commission

4 November 2024

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SUMMARY OBSERVATIONS

CNSC commissioners are being asked to approve the Ontario Power Generation (OPG) application to grant a licence to construct the first ever BWRX-300 reactor on the Darlington reactor site, based on information largely provided by the vendor – GE Hitachi Nuclear Energy (GEH). The vendor has made some dubious claims about this new reactor design that has been successfully sold to OPG.

We are a team of two independent nuclear safety specialists – one a safety engineer with over 44 years of design assessments and safety analyses for BWRs, PWRs, RBMKs and CANDUs in a number of countries, and the other a scientist-mathematician, a nuclear safety consultant for the last 50 years who has spent much of his life analysing nuclear power issues in the public interest.

Based on our examination of relevant documents provided by CNSC, as well as others from GEH, OPG, the US NRC, and others, we perceive serious uncertainties and gaps in many of the claims made by GEH. Indeed, it is admitted by all that the BWRX-300 design is incomplete. Assumptions of major importance have not been substantiated, and central aspects of the safety case remain unproven, untested, and at this point in time, not confidence-inspiring. The discarding of several safety related features that were considered critical in earlier reactor designs is noteworthy.

During the CNSC Licencing Hearing, held on October 2, 2024, Commissioner Hopwood said “It's not entirely clear to what extent the design has been completed in such a way that the conclusions that support a licence to construct are then justified." (Transcript p.105) We concur with the Commissioner's observation, and we go further. We have come to the conclusion that the BWRX-300 reactor design is not sufficiently finalized to allow for the granting of a construction licence at all. Such a move is premature. These hearings are moot. They should not have happened.

It would be irresponsible to grant a construction licence at this stage of development. One can sense the unease that CNSC staff itself feels on this score in its reply to Commissioner Hopwood's implied question:

> "We feel that we, from the benchmarking we've done, where the design is allowed to iterate, the design is allowed to be modified, ***ensuring that you have that completed design prior to construction***…" (Sarah Eaton, CNSC, transcript p.107, emphasis added)

Perhaps without meaning to, Ms. Eaton has agreed with our conclusion, as stated above. She simply says the obvious: that you have to have a completed design prior to construction. The Commission should accept nothing less. Commissioners should insist on a final design that has been thoroughly vetted in the public interest prior to considering the granting of a licence to construct.

1.1 THE REACTOR PRESSURE VESSEL ISOLATION VALVES

The safety case for the reactor design is not finalized in large part because the design itself is unfinished. Some of the most important safety concerns could profoundly alter the construction of the reactor and/or the specification of its component parts. As Commissioner Hopwood went on to say,

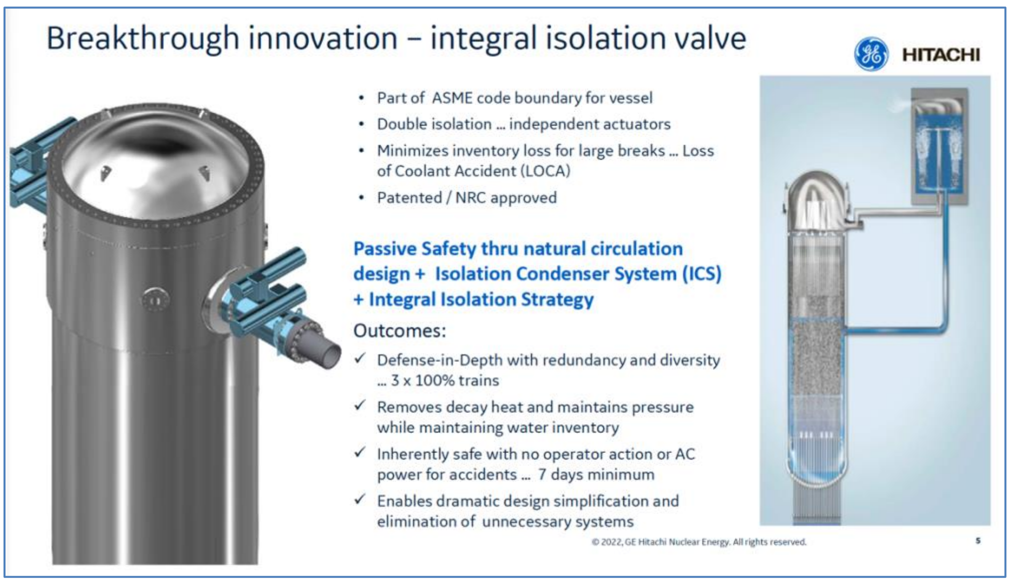
> "So it seems that … you will soon be in a position where equipment is being ordered … the reactor pressure vessel being an obvious case. So could you say

> with confidence that the design is proceeding sufficiently to be able to put a manufacturing spec in place for such key equipment items like that?" (p.109)

David Tyndall of OPG responded:

> "… ***there are things that just don't change***. So when we think about a reactor pressure vessel, we understand the operating conditions of the vessel that we need. We are able to detail that adequately in order to progress …." (p.110, emphasis added)

But Mr. Tyndall's answer is disingenuous. Things ***do*** change. GEH goes out of its way to boast that the BWRX-300 pressure vessel is different from any other nuclear pressure vessel ever built, because it has "integral pressure vessel isolation valves".



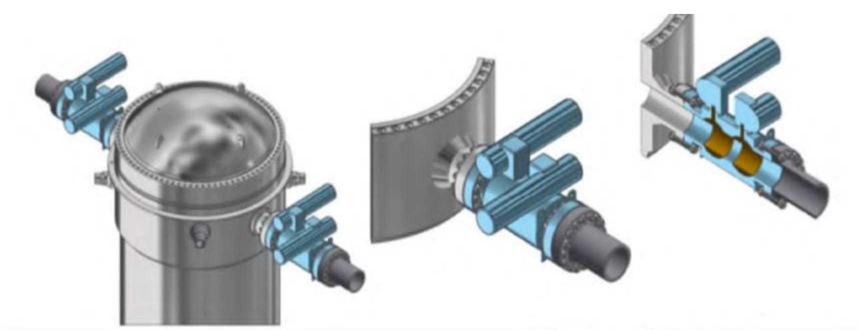


Figure 5.10-1: Reactor Pressure Vessel Isolation Valve Assembly (Example)

These images are from a 2023 BWRX-300 Public Information Presentation.

The existence and the placement of these isolation valves is fundamental to the BWRX-300 safety case. The valves are like taps. They allow the pressure vessel to be sealed off – isolated – from the primary cooling circuit in an emergency. When those taps are closed no water goes in or out.

In the event of a pipe break somewhere in the system, there will be no loss of water from the reactor pressure vessel itself once the isolation valves are closed. The vessel will be cut off – isolated – from the broken pipe. Isolating the vessel eliminates the need for an external ECCS – Emergency Core Cooling System. Up until now, an ECCS has been a critical safety feature for most power reactors. It cools the fuel by continuously adding water to replace what is being lost through the pipe break.

But in the case of the BWRX-300, the isolation condenser system alone (an internal closed loop) is intended to remove the decay heat so as to prevent the core from experiencing severe damage – or even melting – through overheating. See the inset diagram on the previous page. Steam rises. It is condensed back into water and returned to the pressure vessel. In this way no water is lost and yet heat is removed. The nuclear fuel is prevented from overheating and releasing radioactivity.

But it all depends on the valves working exactly as planned. The isolation valves on the reactor vessel have to close, and the valves leading to the "Isolation Condenser System" have to open.

Although GEH and OPG maintain that the isolation valves are "integral" to the reactor vessel one can see that this is not the case. They are not made of the same material; they are in fact bolted onto the reactor vessel, so it is a misnomer to call them "integral" to the reactor vessel. In the staff document CMD 24-H3 (Part 1), on page 5-50, we read:

> "The RIVs are ***connected directly to the reactor vessel using bolted flange connectio*****ns** and are classified as break exclusion areas. All of the RIVs except for those to/from ICS [Isolation Condenser System] are fail-closed type valves." [emphasis added]

"Fail-closed" valves are designed so that when the valves are not in working order they will be closed. A "break exclusion area" is an area where sudden breaks simply do not occur.

As we shall see, it is not correct to say that theses valves are "Break Exclusion Areas".

1.2 VALVES CAN FAIL – "LET ME COUNT THE WAYS"

CNSC staff highlights the vendor's absolute reliance on the reactor pressure vessel isolation valves in the case of a large break Loss of Coolant Accidents (LOCA):

> "during large break LOCAs, ***RPV inventory loss does not threaten fuel integrity. After RPV isolation, decay heat is removed by the ICS*** from the RPV." [CMD 24-H3 (Part 1), emphasis added]

In plain English, when a large pipe break occurs the isolation valves are supposed to snap closed, preventing any water from escaping from the core of the reactor (escaping water is described as an "inventory loss"). Other valves open so that the radioactive decay heat can be removed from the reactor core via a closed loop: hot steam goes upwards and is condensed back into cool water, which is promptly returned to the core of the reactor. In this way the fuel is always submerged in water and cannot overheat. But for this scheme to work, valves have to operate flawlessly.

If the nuclear fuel were not submerged in water, overheating would occur and a chemical reaction would take place between the steam and the metallic cladding that coats the fuel rods. The oxygen (O) from the water molecules (H_2O) will combine with the metal to produce an oxide, liberating the hydrogen (H_2) which is a very explosive gas. Hydrogen recombiners are normally used to "burn" the hydrogen gas, producing water molecules – combining the hydrogen with oxygen to produce H_2O.

But the BWRX-300 design has no hydrogen recombiners and no Emergency Core Cooling System because the isolation of the reactor vessel and the heat removal peformed by the Isolation Condenser System never leaves the nuclear fuel uncovered with water. The system is regarded as foolproof.

But there is no such thing as a foolproof system, because the fool is always greater than the proof. The effectiveness of the ICS [Isolation Condenser System] to remove the decay heat without uncovering the fuel (thereby not generating hydrogen gas) depends on the flawless performance of the RIVs [reactor isolation valves] and the ICS. One set of valves has to close, and another set of valves has to open. If either set of valves fails badly, all bets are off. Where then is the backup? What's Plan B?

For example, if just one of those isolation valves on the reactor vessel fails to close in the event of a large break LOCA [Loss of Coolant Accident] the reactor coolant boundary will be breached. Water will rush out of the reactor vessel and be lost through the pipe break. The water level in the core will drop, potentially uncovering the fuel. Severe fuel damage is next, and that means large radioactive releases.

Such a scenario puts an enormous degree of importance on the RIV isolation valves. They absolutely must close when called upon in order to preserve the reactor coolant boundary integrity and maintain the water level inside the core to keep the fuel covered with water within the reactor vessel.

Although the RIV valves on the BWRX-300 reactor vessel are designed to "fail closed", it should be recalled that the Three Mile Island (TMI) meltdown in 1979 in Harrisburg Pennsylvania was caused by the failure of a pressure relief valve to automatically close, as it should have done, when the pressure dropped. But it didn't. It stuck in the open position. It was supposed to be a "fail-closed" valve. It wasn't.

The sequence of events at TMI is simple enough to understand in retrospect. Due to one manually shut valve [a human error], one of the TMI steam generators was deprived of feedwater for a couple of minutes. As a result, pressure built up rapidly and a pressure relief valve on the primary coolant side automatically opened to prevent overpressurization of the primary heat transport system. That relief valve should have closed automatically when feedwater was restored to the steam generator shortly afterwards. But it didn't. It took two days for TMI operators to find the stuck-open valve, during which time the core was uncovered and fuel melting occurred. The relief valve that was stuck open acted like a small pipe break that insidiously drained the core of coolant, and the fuel ended up being uncovered.

So the BWRX-300 safety strategy puts an enormous degree of faith in valves, especially in the reactor vessel isolation valves. They are attached to the reactor vessel by means of a bolted flange connection. Is that connection foolproof?

Here is a quick run down on the nature and complexity of a bolted flange connection such as that used to affix the RIV isolation valves to the BWRX-300 reactor vessel:

> "A bolted flange connection is a complex mechanical system whose components must be selected and assembled properly to provide reliable sealing over a wide range of operating conditions. All of the various components of the assembled bolted flange connection are important to the proper operation of the joint. The components consist of the piping, or vessels, the flange(s), the gasket(s) and bolts. In addition to the components themselves, the joint design and assembly are critical to the long-term operation of the joint."
>
> "Once seated, a [bolted flange connection] gasket must be capable of overcoming minor alignment issues, flange sealing face imperfections and operating variations such as, but not limited to:

- Non-parallel flange faces
- Misaligned flanges (Figure 3)
- Distortions, troughs, or grooves
- Surface waviness (deviation) (Figure 4)
- Surface scorings (Figure 5)
- Other surface imperfections (Figure 6)
- Flange deformation or warping of the flanges (ex: bowing) from applied forces on the flanges. Often this is due to incorrect gasket selection and may result in failure.
- Allowable temperature range relative to the equipment and/or system.
- Allowable pressure relative to the equipment or system.
- Ambient conditions such as outside temperature or outside elements such as chemicals that may come into contact with the gasket.
- Startup and shutdown processing variations
- Hydro test pressure during leak testing
- System internal cleaning/flushes"

From A.R. Thomson Group Inc., "Forces Acting on the Bolted Flange Connection"

https://www.arthomson.com/forces-acting-on-the-bolted-flange-connection#:~:text=A%20bolted%20flange%20connection%20is,proper%20operation%20of%20the%20joint

It must be borne in mind that the expected lifetime of the RIV isolation valves is 60 years – the expected lifetime of the BWRX-300 reactor. It is necessary for the Commissioners and the CNSC staff to be fully cognizant of the critical importance for safety that GEH has placed on the flawless functioning of these valves. Until a full independent safety analysis has been carried out on this matter and approved by the Commissioners, no construction licence should be granted.

See Annex A, "Introduction to the common failure modes of valves". Of particular relevance is the discussion on the final page, "What are the most common reasons for valve failure in pressure vessels?" That discussion about pressure vessels appears after a more general discussion of the causes and effects of the most common modes of valve failure: Leakage (external and internal), Sticking or Binding, Erosion, Cavitation, Corrosion, Stem Failure, Packing Failure, Actuator Failure, Thermal (expansion or contraction), Galling, Over-Pressurization, Elastomer Degradation, Seat Wear or Damage, Improper Installation, Material Incompatibility, Design Flaws, Inadequate Testing, Maintenance Overlook, Improper Sizing, Faulty Actuation Mechanisms, Environmental Factors, and Operational Errors.

If any changes or adjustments to the BWRX-300 design of the reactor vessel (with its attached isolation valves) are to be made in the interest of public safety, no construction licence should be issued until such changes have been decided on, fully analyzed with appropriate backup strategies, and approved by the Commissioners.

1.3 THE BREAK EXCLUSON ZONE (BEZ) – 2024 MEETING

One of the reasons for the vendor to describe the RIV isolation valves as "integral" to the reactor vessel (which we maintain is simply not true) may be to allow GEH to more easily include those valves in what they call the Break Exclusion Zone (BEZ). In that zone, which according to the vendor includes everything inside the steel-lined containment structure, GEH asserts that no breaks can occur without ample prior warning. That allows for operator action such as reactor shutdown if necessary. The BEZ concept is based on the principle of Leak Before Break (LBB), a characteristic that has to be established through evidence based reasoning.

It has long been accepted by the US Nuclear Regulatory Commission (NRC) that the failure of a reactor pressure vessel is not a credible event. Therefore, if the isolation valves are "integral" to the reactor vessel, they may also be regarded as de facto break-proof and perhaps even invulnerable to failure. This of course would be an unwarranted conclusion. Valves can fail. And valves can break!

At the October 2 Licencing Hearing (Part 1) CNSC staff openly admitted their lack of familiarity with the BEZ concept. On page 68, Bartek Rzenkowski states:

> CNSC staff note that the BEZ concept is not addressed in Canadian nuclear regulatory framework and it is not a standard practice in the Canadian nuclear industry. OPG will be required to provide technical information supporting the use of this concept at the extent of the proposed BEZ implementation, as outlined in CMD 24-H3 Appendix D.2.

CNSC's unfamiliarity with the BEZ concept and its implications for the overall reactor safety case of the BWRX-300 has apparently not been adequately dispelled in the eight months since CNSC staff participated in a public meeting conducted by US NRC staff on January 24, 2024, focussed precisely on the vendor's use of the BEZ concept (see subject line below):

From:	Jim Shea
Sent:	Thursday, February 8, 2024 6:50 AM
To:	GEH-BWRX-300MtgSumPEm Resource
Cc:	Jordan Glisan
Subject:	January 24, 2024 Meeting Summary regarding the GEH approach to the BEZ for the BWRX-300
Attachments:	January 24, 2024 Public Closed Meeting Summary on NRC Staff Feedback on GEH BEZ for BWRX-300.docx; BTP 3-3 & 3-4 Public Meeting Slides.pdf

On January 24, 2024, from 10:00am to 11:30am, the NRC held an open and partially closed meeting with GEH regarding the application of NRC SRP BTP 3-3 & 3-4 to the piping configuration associated with the BWRX-300 SMR design.

The public / partially closed meeting summary is attached along with the GEH presentation and piping configuration that is the subject of the meeting.

If there are any further questions regarding this subject, please contact the NRC project manager.

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james.shea@nrc.gov
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Mail Stop O-7D21
Washington, DC, 20555-0001

The US NRC's Public Meeting Summary for this January 2024 event is appended to our report as Annex B. That document also includes, as an enclosure, the vendor's "Presentation Slides for Pre-Application Meeting for GE Hitachi Nuclear Energy BWRX-300 Proposed Break Exclusion Zone Design Requirements". The NRC summary acknowledges that "The Canadian Nuclear Safety Commission (CNSC) review team attended this meeting remotely, as part of a 2019 memorandum of cooperation between the NRC and CNSC, and a September 2022 "Charter – Collaboration on GEH's BWRX-300 Design."

The report shows that NRC licensing staff agrees with our own overall conclusion, stated earlier, that it is premature and inappropriate to grant a licence to construct based on the incomplete nature of the BWRX-300 design. Even GEH describes its January 2024 event as a "Pre-application" meeting. The NRC Summary says:

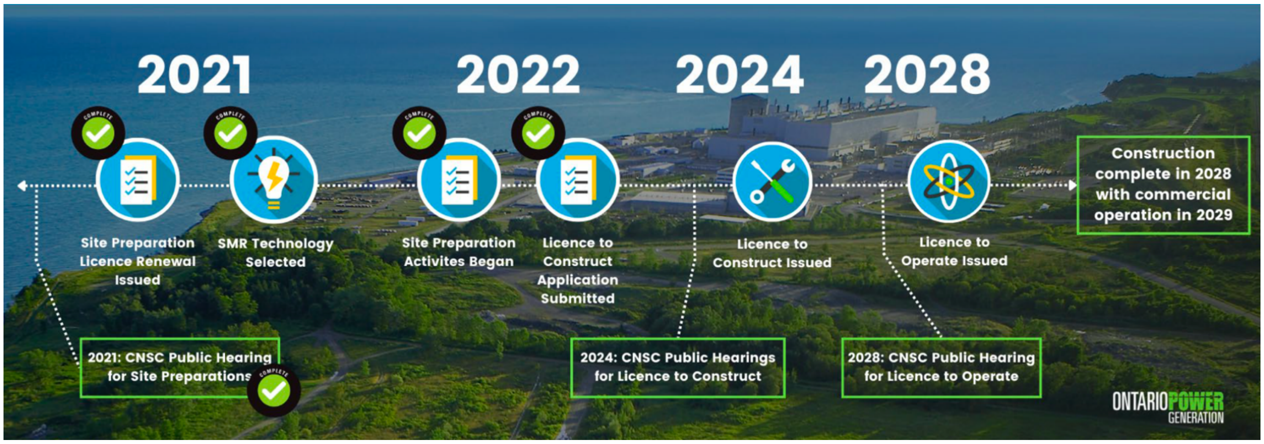
> "GEH stated, 'Design details are to be described during future licensing activities' and the staff noted that, 'Specific aspects of ***the connection of the RPV isolation valves to***

> ***the reactor vessel*** will be reviewed during future licensing activities of the BWRX-300 SMR.'" [emlhasis added]
>
> "GEH indicated in the submittal that the detailed design of the BWRX-300 SMR including ***the piping configuration inside and outside containment is not complete*** therefore, the NRC staff will make ***a final determination of the BWRX-300 SMR's acceptability when the detailed design is completed and reviewed*** by the NRC staff during future licensing activities." [emphasis added]

Here we see that, as of calendar year 2024, NRC is not entirely at ease with the manner in which the isolation valves are attached to the reactor vessel. It is also clear that until the detailed design of the reactor is completed, NRC will not make a final determination of the acceptability of the BWRX-300 design.

In light of Canada's total lack of experience in licensing any power reactor designs other than CANDUs, combined with staff's admitted unfamiliarity with the BEZ concept, mindful of CNSC's cooperation agreement and charter of collaboration with NRC, as well as the incomplete nature of the vendor's piping configuration inside and outside containment for the BWRX-300 design, it is distressing to see CNSC staff urging the Commissioners to grant a construction licence at this very early stage. This behaviour is not commensurate with the enormous dangers that millions of Canadians in the Oshawa area may face if the safety of the reactor proves to be wanting in any serious respect.

Could it be that the corporate timetable laid down by the proponent OPG, is receiving undue consideration by staff at the expense of the only legally mandated role of the CNSC – to protect the health and safety of Canadians and the environment? If this is not the case, then why the rush? Is it acceptable that the proponent should dictate to CNSC when a licence to construct should be granted?



1.4 SAFETY STRATEGY FOR THE BWRX-300 – 2022 MEETING

In a December 2022, CNSC staff participated in a public meeting hosted by US NRC staff on the Safety Strategy for the BWRX-300. The meeting summary is attached to this report as Annex C. In that report we learn that

> "GEH specifically stated during a previous public meeting dated June 29, 2022, that ***after implementation of its final design and submitting it for staff's review*** … the BWRX-300 would expect to meet all the applicable NRC regulations and guidance as well as meet the design requirements in CNSC REGDOC-2.5.2." [emphasis added]

Once again we see that it is only "after the implementation of its final design" that the requirements in CNSC REGDOC-2.5.2 can possibly be met. Even the vendor GEH admits that. And CNSC staff know it. After all, they participated in the 2022 NRC meeting. Why then are they pushing for these licencing hearings to be held when they know that it is premature? That all the answers are not yet known? If the Commissioners wish to show integrity, they should close these hearings down and reprimand staff for initiating hearings before the essential safety facts are available.

The summary report of the 2022 meeting (Annex C) states that

> "CNSC staff listened to the discussions during the meeting and asked clarifying questions. At that time, CNSC assessment of GEH's proposed Safety Strategy (as described in the White Paper) consisted of feedback to GEH that ***the proposed strategy appeared to be generally consistent with CNSC's regulations and processes***; and that they would give a more detailed assessment at the scheduled follow-up meeting."
>
> "However, the NRC staff's assessment of the BWRX-300 Safety Strategy (as described in the White Paper) consisted of feedback that ***GEH should provide additional information to adequately address all elements of the NRC's regulatory framework*** including risk informed performance-based decision making which is based on regulatory compliance, maintenance of safety margin, and treatment of uncertainties. The NRC staff also reiterated that while this novel approach by GEH could be successful, NRC approval of the safety strategy will be based on an applicant showing conformance to NRC regulations or justifying applicable exemptions." [emphasis added]

During the meeting, NRC staff raised a number of important challenges to GEH regarding the BWRX-300 Safety Strategy, as spelled out in Annex C :

(1) GEH's novel approach might work but it must conform to NRC regulations.
(2) Every LOCA must be analyzed regardless of low frequency estimates
(3) Steam line rupture and rod drop accidents must be analyzed as design basis events
(4) Quantitative calculations of severe core damage and large release frequency are required
(5) Demarcation between design basis events and design extension conditions must be clear
(6) Treatment of design basis hurricanes, hurricane missiles, and tornadoes is incomplete
(7) Mitigation of beyond-design-basis-accidents must be addressed
(8) "Practical elimination of large releases" should be re-examined

Canadians deserve better from their nuclear regulator. Why have intervenors never had the opportunity of seeing CNSC staff interact with potential licensees in pre-application public hearings, as NRC staff does, where proponents are challenged to meet demanding regulatory conditions and answer difficult question about their assumptions and their analysis? Why is CNSC staff so willing to accept the claims and assumptions made by proponents without challenging them to back up those claims and substantiate those assumptions? Such passive behaviour on the part of CNSC staff makes the Canadian nuclear regulator seem more like a lapdog than a watchdog. What evidence do we have that staff has ever held the proponent's feet to the fire on crucial safety issues related to this new, never-before-built reactor?

1.5 CONSEQUENCES OF A STEAM LINE RUPTURE AND WORSE

It is well known that the sudden break of a steam line can be extremely damaging. Decades ago, it was discovered by CNSC staff that if a steam line passing directly over the control room at the Gentilly-2 reactor ruptured, it would kill everyone inside the control room. Instead of relocating that steam line, a compromise was worked out with Hydro Quebec to closely monitor the integrity of the steam line using a leak-before-break (LBB) approach. In a similar way, GEH wishes to use an LBB approach to rule out the need for them to consider any sudden ruptures inside the containment structure of the BWRX-300.

In the NRC Summary of the BEZ meeting in January 2024 (Annex A), NRC staff observes that

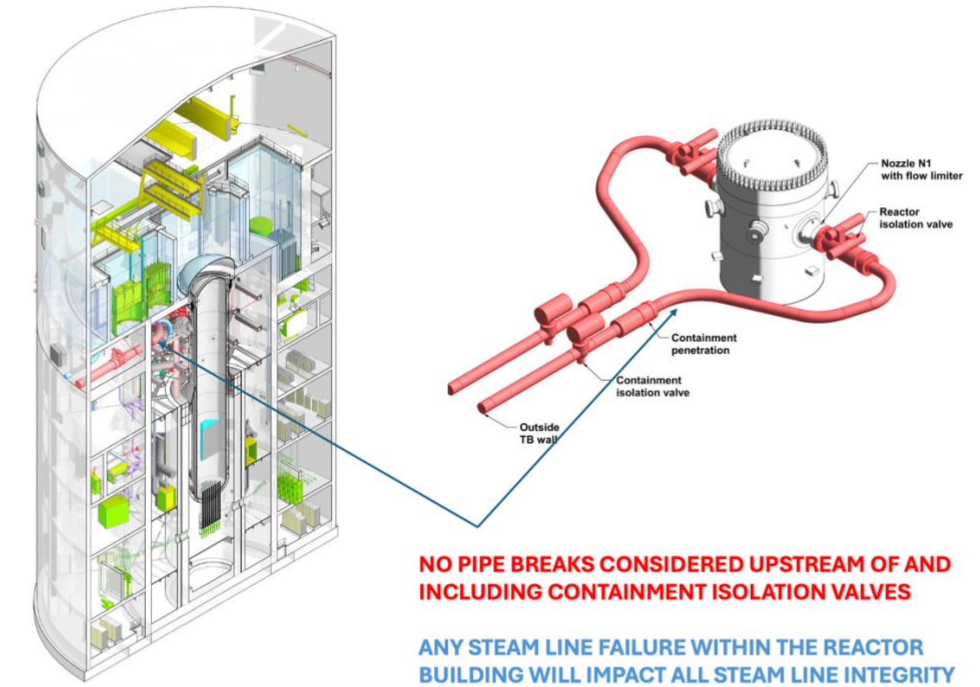
> "A staff-approved leak-before-break (LBB) analysis permits licensees to remove protective hardware such as pipe whip restraints and jet impingement barriers, redesign pipe connected components, their supports and their internals, and other related changes in operating plants…. Likewise, requirements for plants under construction or being designed are similarly relaxed."

However,

> "LBB is applicable only to an entire piping system or analyzable portion thereof. LBB cannot be applied to individual welded joints or other discrete locations. Analyzable portions are typically segments located between piping anchor points."

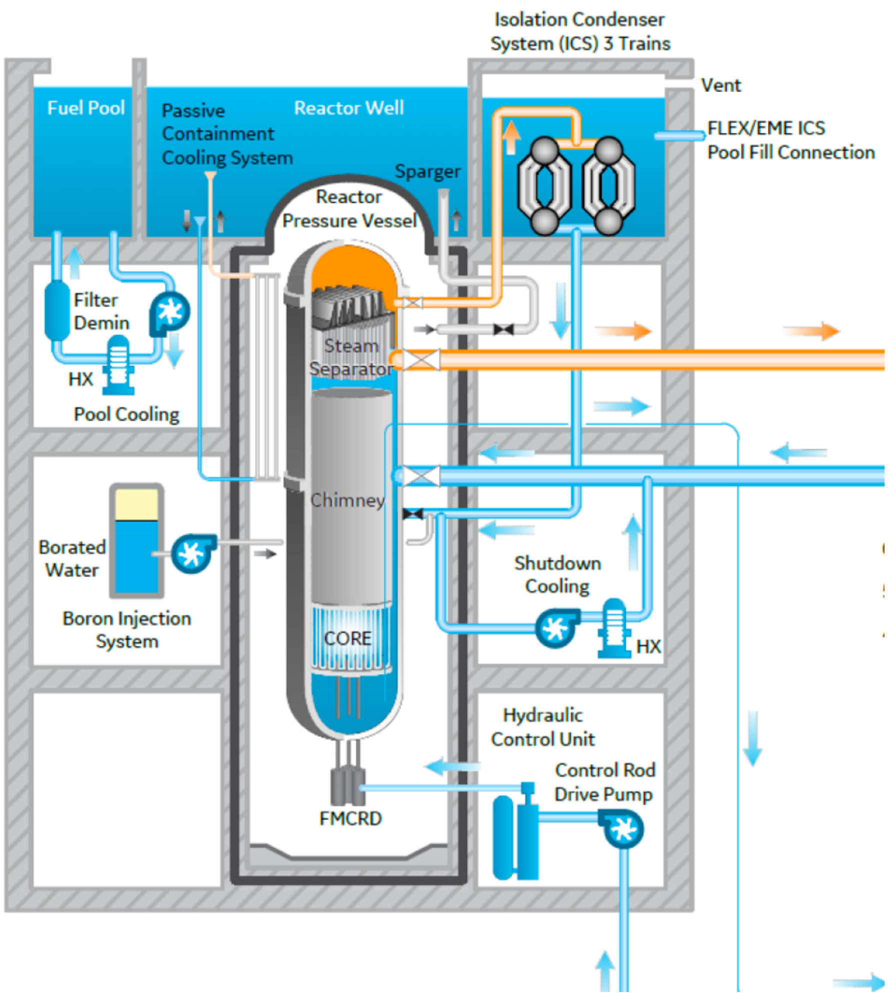
This puts to rest the GEH claim, quoted earlier, that the pressure vessel isolation valves are by definition a "break exclusion area". That is untrue. Since the entire piping system is not yet described in sufficient detail, NRC staff are adamant that determinations of LBB and BEZ will have to be made later. In order to satisfy the extremely low probability criterion, NRC staff states that it will need

> "a deterministic evaluation of the piping system that ***demonstrates sufficient margins against failure, including verified design and fabrication and an adequate in-service inspection program, including leakage detection capabilities***." [emphasis added]



Both the spent fuel pool and the pools of water needed by the isolation condenser system are on the ground level, even with the top of the reactor vessel. Under adverse circumstances, a violent steam line rupture could incapacitate the isolation condenser system, and possibly even jeopardize the integrity of the spent fuel pool and/or its cooling system. During the 2022 meeting, NRC staff pointed out to GEH the absence of mitigation measures "related to the Spent Fuel Pool level monitoring and cooling makeup capabilities."

The destruction or draining of the Spent Fuel Pool could result in the overheating and unfiltered dispersal of several years worth of high-level radioactive waste, creating an unimaginably large radioactive release.



This diagram is taken from page 24 of GEH's 2019 Status Report on BWRX-300.

It is not clear why the Spent Fuel Pool has to be located in such a vulnerable position. The safety implications of leaving it where it is or moving it to a more sheltered location with hardened shielding should be considered before any construction licence is granted.

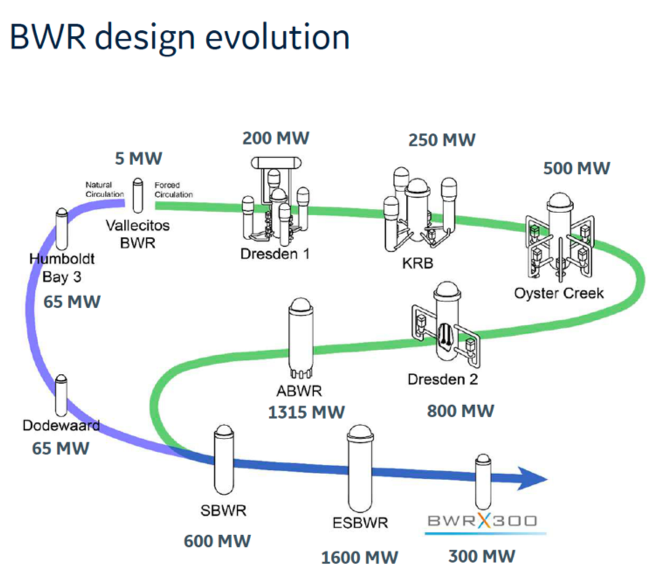
Similarly, destruction or incapacitation of the ICS would eliminate the only method available for removing decay heat from the core, since the BWRX-300 design has excluded all other emergency cooling systems that might serve this purpose. Again, the implications of such vulnerabiliy should be weighed carefully and mitigation measures be clearly articulated before a licence to construct is considered.

It is unclear to what extent the top of the reactor vessel is within containment, but certainly the isolation condenser system and the spent fuel pool are not. Both – and possible all three – would be in danger of total devastation in the event of a large airplane crash directed towards the top of the reactor well. According to REGDOC 1-1-2-v2 an application for a licence to construct a nuclear reactor [emphasis added]

> "should describe the analysis of all potential hazards (internal and external), both natural and human-induced. Some examples are:
>
> - for natural external hazards: earthquakes, droughts, floods, high winds, tornadoes, abnormal surges in water level and extreme meteorological conditions
> - for human-induced external hazards: those that are identified in the site evaluation, such as ***airplane crashes*** and ship collisions
> - for internal hazards: internal fires, internal floods, turbine missiles, onsite transportation accidents, and releases of hazardous substances from onsite storage facilities"

1.6 THE EVOLUTION OF THE BWRX-300 DESIGN

The vendor explains that BWRX-300 is the tenth phase in the evolution of the Boiling Water Reactor GEH family.



We agree with the applicant that the BWRX-300 reactor design is based on a number of similarly fueled boiling water reactor cores and draws some inspiration from a couple of 'advanced' BWR reactor designs., In spite of a number of unfortunate events in the earlier versions of BWRs by GE, including short lived operations and sudden shutdowns and – of special note – the Fukushima Daiichi fiasco, we recognize that the proposed design has been presented as a new 'innovative' BWR.

However, weighing serious impartial reactor safety considerations and the national interest, this 'tenth' generation GE BWR design is certainly far from ready for a license to begin construction. Already there is a billion dollar federal commitment, but the design remains incomplete. There is a huge public safety risk associated with operation as is, and a large liability for waste & decommissioning – timely or otherwise.

Of course, the first question that comes to mind is the wisdom in choosing the proposed location, a mere 300 odd meters from an existing multi-unit station, in complete disregard of siting guidelines followed by the CNSC's partner, the US NRC, in reference 1. Then we have a bizarre design that includes placement of the reactor over 100 feet below grade and waterline, about 100 metres from a lake used by millions.

We attach with our submission disturbing information (references 2, 3) by a credible witness on another similarly below grade GE boiling water reactor at Humboldt Bay, California. The plant was touted by OPG staff, in our last meeting with them in October 2024, as a GE 'success story'. Built at a cost of about US $60 million, it suffered a hair-raising accident in 1970 and has already cost the California PG&E utility $1600 million to partly decommission it (but spent fuel casks still remain on site 54 years later).

In 1970, fuel damage occurred at the Humboldt Bay plant. The isolation condenser system failed because a valve could not be opened. In 2011, the isolation condenser system at another GE BWR – Fukushima Daiidhi unit 1 – operated adequately for some hours after the earthquake but eventually failed, for unexplained reasons, soon after the tsunami struck. The reactor was about 40 years old at the time.

This submission is a sincere attempt to implore the Commission not to "rubber stamp" an industry request to proceed prematurely. CNSC's legal and moral obligation is to the Canadian public, not to the nuclear industry – and not to the government either. Canada cannot afford to take a chance on a nuclear disaster or gamble with public safety. CNSC must live up to its motto: "We will never compromise public safety.".

We recognize glaring holes in the safety story of the BWRX-300 from lives spent doing design and safety assessments not only for CANDUs but also BWRs, PWRs, RBMKs – all of which have had design and/or operational failures that not only caused a trillion dollars in damages but also crippled severely public trust in the industry. So, ignoring the euphoric idea of saving the world with quick to assemble SMRs, we have taken a look under the hood and found some very disturbing lapses in best practice. There are a large number of safety issues apparent to us in spite of the very limited amount of design information that has been made available to us. Most of it is restricted or withheld from public access.

The proposed BWRx-300 is a very odd simplification, a kind of deviant. It differs from the ESBWR design not only in size but also in safety functions. Note that the ESBWR received a final design Approval from US NRC staff in 2011, but that was before the full implications of the triple meltdown of Fukushima Daiichici had beome evident. ESBWR was never built. It was retired – by GEH request to the NRC – in 2014.

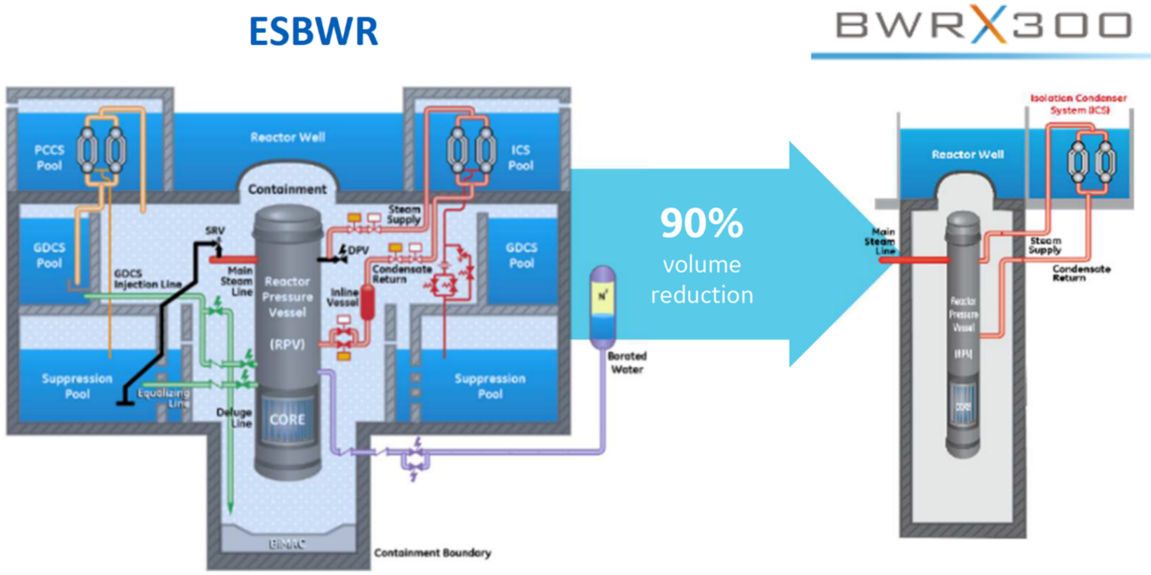


Figure 1 : In a rush to be small, some sound safety elements from ESBWR were dropped. Little similarity remains between the two.

The BWRX-300 design presented for consideration by the Commission is so obviously lacking in basic safety features (see Figure 1) and so heavily burdened by a multitude of unsubstantiated claims (some of which we discuss in this submission) that it is unfathomable that any CNSC staff cognizant of reactor safety requirements, would recommend that you, the Commissioners, issue a license to begin construction. That is a process that, looking at recent pictures of the proposed site, goes well beyond what would be a 'site preparation' and thus seems to have already been started, nevertheless.

It is the responsibility of the Commission members to seriously take advice from all available sources, including intervenors and senior members of the nuclear community like us. You will see that not only does the OPG application fail to make a convincing case, the CNSC staff seem only to have nodded agreement and at times have made unsubstantiated claims and statements that are unfortunate and unwarranted for an agency of the government.

While trying to claim safety features in a section titled "Unique **Design Features",** the recent BWRX-300 design description in reference 4 by OPG actually lists the greatest safety deficiencies in the proposed reactor. The design deficiencies are addressed in the following subsections and include, but are not limited to:

- No overpressure protection by pressure relief valves on the primary coolant circuit / reactor pressure vessel
- No overpressure protection by pressure relief valves on the steel-lined containment.
- No Emergency Core Cooling System (ECCS) at all
- No emergency AC power by diesel generators or otherwise
- A containment 10 times smaller in volume for a reactor that is 1/2 to 1/3 the power
- Over-dependence on the triple loops of the Isolation condenser system

Missing in the design are some very fundamental features that are basic to safety, engineered systems like LOCA mitigation by ECC, containment integrity control by pressure suppression systems and hydrogen control for design basis accidents and a number of its claimed predecessor ESBWR's systems that supported that design's defense in depth claims. As a result, risk to the public from operation of the BWRX-300 so close to the heart of Canada is very high and should not be allowed to proceed. In addition, a construction workforce of over a thousand will be in jeopardy from Darlington whose severe accident mitigation measures are abysmally inadequate and whose risk profile to severe core damage has been misrepresented by both the OPG and CNSC staff. One day this might cause severe damage to both organizations (see Reference 5) . Any construction permit application approval by the Commission members will be contrary to their legislated obligations and public interest.

We are also cognizant of the Commission member and Commission staff perceived obligations to their past and future employers. So that the best we can really expect is the least we know you must do – delay the decision until a redesign is worthy of an application and then find staff for an honest design evaluation and an unbiased recommendation for approval.

There also are clear statements by the US NRC that the current design may require substantial changes such as a significantly larger / enlarged containment , additional emergency core cooling and overpressure protection measures that will make any premature construction activity an irrecoverable error.

It is hoped that the Commission members will hold public safety paramount to deny or delay the approval. While it is so obvious why the request should be denied and if in keeping with the long CNSC tradition of giving in to industry is continued, it will be long regretted by all parties.

With simple but critical design parameters hidden in the name of 'proprietary' information from independent nuclear safety advocates like us, there are the usual indications that the decision to engage in public and other stakeholder consultations is perfunctory; without a realization by the concerned parties that the OPG, CNSC staff and Commission personnel who knowingly do so, can likely be held liable personally just as were (initially) the TEPCO officials who ignored reactor vulnerabilities of the three Fukushima BWR/s that melted down following the 2011 accident.

Please note that in the current case the applicant has invoked some very unusual claims of absolute safety, presenting for our consideration a reactor design that has none of the backup safety features normally required in a responsible defense-in-depth approach. Claims of compliance with the US 10 CFR 50 requirements are largely misleading. For example it is claimed that the conditions are satisfied for a LOCA inside the containment, without clearly stating that a LOCA in containment is ruled out arbitrarily by invoking the "Break Exclsuion Zone" concept :

> "10 CFR 50 Appendix A, GDC 50 – BWRX-300 containment [is] designed to withstand calculated pressure and temperature conditions resulting from LOCA with sufficient margin."

There are also unmet US NRC requirements re severe accident mitigation as laid out in Reference 6.

1.7 HAS CNSC OR OPG DONE ANY TECHNICAL ASSESSMENT OF BWRX-300?

The Commission members should inquire if there is any independent design or safety assessment by CNSC staff to critically assess the vendor claimed 'Safety Strategy' and have declared in multiple meetings with NRC that "the proposed strategy appeared to be generally consistent with CNSC's regulations and processes" while the NRC staff in contrast have stated in various forums that the GEH strategy proposed "may not align with the requirements for structures systems and components (SSCs) under 10 CFR Part 50 or 52"; " This proposed approach has the potential to result in the final BWRX-300 design which is not in compliance with NRC regulations".

Reason is very simple. CNSC does not have the requisite trained, impartial technical personnel to do the job of doing safety assessments on a new reactor design, having failed noticeably in applying lessons learned from Fukushima to the one reactor design they have spent 50 years giving a pass on and missing out on the design deficiencies in the operating plants that are clearly stated in the safety reports but implications ignored. As an agency in discernable collusion with the industry, the CNSC staff recommendation to approve a construction license to a reactor that has not yet been designed beyond the initial concept, is incomprehensible. It is also strange that the staff seem to have given no technical presentations in their joint meetings with the NRC staff.

CNSC staff has unilaterally declared that their containment requirements are largely technology neutral. It has made bizarre statements in reference [7] like:

In some novel designs, the expected contribution of the different barriers of the containment system to the fulfilment of the safety functions is different than for traditional reactors. In these new designs, fuel may be considered the dominant contributor to the confinement function, and less importance may be placed on the containment structure.

1.8 PROPOSED BWRX-300 REACTOR IS NOT A SMALL REACTOR

The BWRX-300 is really a medium sized nuclear power reactor with reactor power greater than that of over 100 nuclear reactors that have ever operated and of 22 that are still operating (see data and pictures in Appendix 1) without any of them ever making claims of being 'small'.

Its footprint has been made small by coming up with a very tenuous argument about an ability to exclude from consideration any Loss of Coolant Accidents within the containment and thus making the containment structure dangerously small (see section below on the claimed Break Exclusion Zone).

By the way, the argument about being a saviour for the environment is certainly untrue. The 240 fuel assemblies with 324 kg/assembly has a total mass of 324 x 240 = 77.76 tonnes with 2.7% average enrichment. Thus the fuel rerquirement for this 300 MWe reactor would need about 4 times more natural uranium mined, processed, and enriched than a 660 MWe CANDU which has a uranium fuel mass of about 91 tons (20 kg of natural uranium of 0.7% U-235 in each of its 4560 fuel bundles).

1.9 REACTOR CONTAINMENT IS DANGEROUSLY TOO SMALL

A primary containment free volume of 5,600 m^3 (Reference 8) is about 5 times smaller than most likely needed for such a reactor (~870 MWth) to mitigate overpressure failures that cannot be excluded once a feedwater or steam line fails within the containment and the pipe whip disables the isolation valves. The argument about the Containment isolation valves being 'integral to the reactor pressure vessel' and hence infallible, ignores so many modes of valve failure, especially of one constantly in path of hot, high velocity fluids, under structural stresses, corrosive environments, undergoing erosion and never subjected to any interior inspections. Valve failure data are abundant.

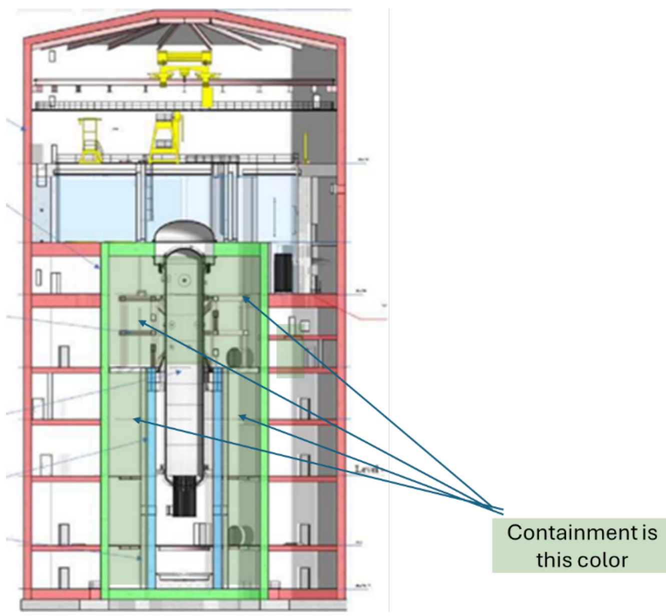


Figure 2: the small reactor containment in BWRX-300

Here is a vendor rendition of the isolation valve:

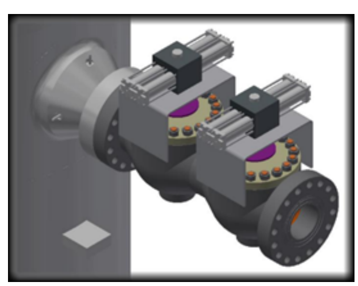


Figure 3 : Rendition of an isolation valve, more than 20 of various sizes to be used in BWRX-300

Given the various modes by which flanged connections have failed due to many reasons, including vibrations, there is no engineering basis to calling them 'integral' to the reactor pressure vessel. That is contrary to common sense. A finite failure probability exists, especially due to the large structural loads on these valves.

Here is an illustration of the BWRX-300 containment volume as compared to other reactors (base figure courtesy of Union of Concerned Scientists, reference 9) :

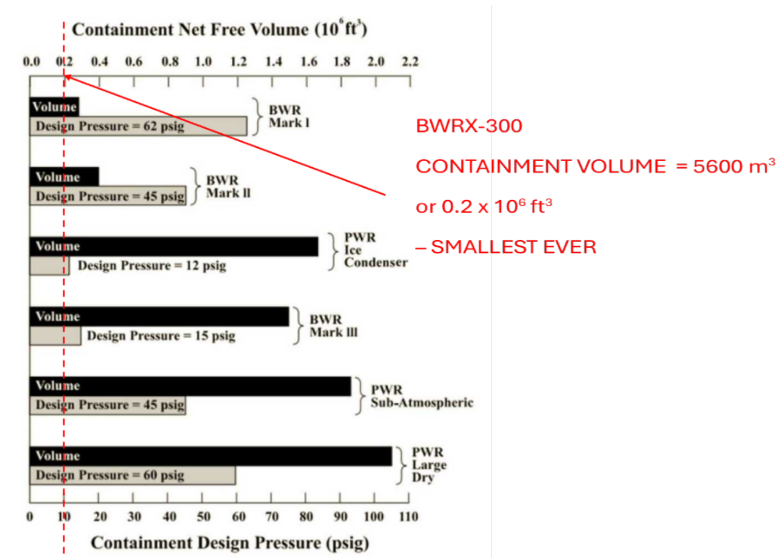


Figure 4: Relative containment volumes and design pressure of various reactor containments.

Containment overpressure mitigation is a fundamental containment design feature that is missing in the BWRX-300. In other Boiling water reactors, it is provided by a pressure suppression system as illustrated in Figure 5 below.

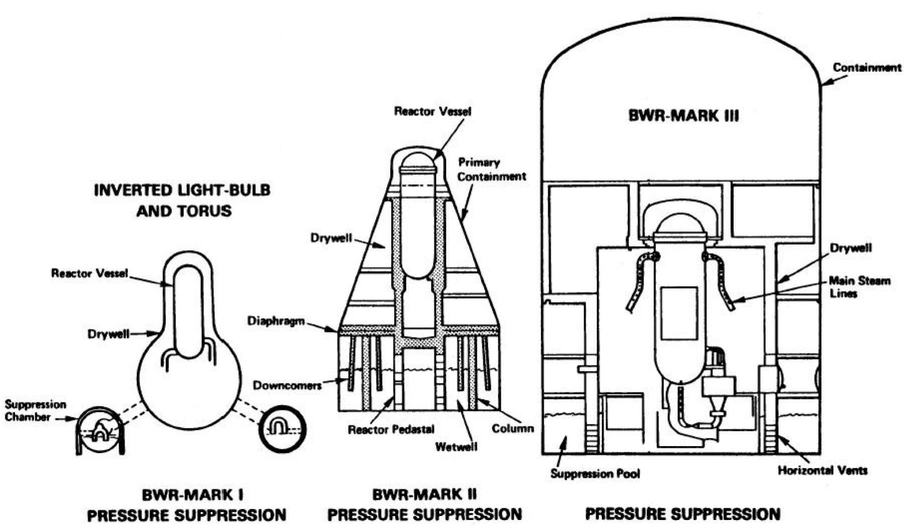


Figure 5 : Different BWR methods of pressure suppression in containment

1.10 THERE IS NOTHING MODULAR ABOUT BWRX-300

As a first of its kind, the proposed reactor has nothing except fuel assembly design (actual height not disclosed) and steam turbines that are common with other reactors and unless there were multiple projects in parallel, there is not a thing that is expected to be "modular". Visions of the reactor being assembled on site by 'modules' trucked in on 18-wheelers are wrong.

1.11 BWRX-300 IS NOT A NATURAL CIRCULATION REACTOR

A claim of the BWRX-3—reactor being a natural circulation reactor is also surreptitious. That implies to many that the BWRX-300 heat transport system or heat removal under shutdown conditions works without pumps. The only natural circulation the BWRX-300 sports is the one inside the reactor vessel – feedwater circulating to bottom of the fuel bundle - just like all PWR boilers – by gravity, not design. Natural circulation through the isolation condenser after reactor shutdown is another matter of concern if effect of non-condensable and valve reliability is included. The shutdown cooling seems to require a pump as shown below:

1.12 LACK OF REDUNDANCY – LACK OF BACKUP SYSTEMS

According to safety norms and IAEA guidelines. "Redundant heat removal systems should be provided to the extent necessary to enable controlled cooldown of the reactor coolant system when the ultimate heat sink is not available, or the main steam line is isolated." Although redundancy may seem to some as abstract and esoteric, it is a foundation stone of nuclear safety as the punishments from ignoring this principle are astronomic. In the case of deployment of a new technology or dependence on a loudly proclaimed confidence that the isolation valves cannot fail to act as desired one must consider that we cannot afford an epistemic accident just because a loudly proclaimed technical assumptions of BWRX-300 design (Isolation condensers are sufficient and dependable; Vessel and containment isolation valves cannot fail; in-containment LOCA cannot be large; pipe whip cannot damage isolation valves, no hydrogen explosions are likely because of the inerting of containment atmosphere by Nitrogen, etc.) proves to be erroneous even if there were seemingly reasonable and logical reasons when these arguments take hold at the designer or their public relations arm.

1.13 NO SAFETY RELIEF VALVES FOR OVERPRESSURE PROTECTION

See Section 1.61 "Enhancements in Safety System Design" of reference 10 for the interesting statement that essentially says that in this 870 MW BWR no safety relief valves for overpressure protection of the billion-dollar reactor are required. The Isolation condenser can take over that role for the containment, we are told. Go figure.

> *The large capacity Isolation Condenser System (ICS) provides overpressure protection in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, Class 1 equipment. Historically on BWRs, the safety relief valve*

> *inadvertent actuation has been the most likely cause of a LOCA and have, therefore, been eliminated from the BWRX-300 design.*

!!!

Just add to that the statement in reference 11 that the condensate return valves are designed to fail As-is, meaning that they can fail closed and since they are normally closed, can fail to open and disable the oh so important Isolation Condenser return.

Please note: The NRC staff will evaluate the compliance of the final design of the RPV isolation and overpressure protection features for the BWRX-300 SMR during future licensing activities in accordance with 10 CFR Part 50, "Domestic licensing of production and utilization facilities," or 10 CFR Part 52, "Licenses, certifications, and approvals for nuclear power plants," as applicable.

1.14 NO EMERGENCY DIESEL GENERATORS

This is also an unfathomable design error

It is stated in reference 10:

> *Elimination of active emergency core cooling systems eliminates the need for onsite emergency power systems. Standby diesel generators are provided for asset protection only.*

This is an incomprehensible safety lapse. There is no safety-related power supply other than limited Class 1E battery-stored power (Reference 12).

There are so many reactor systems that rely on power that must be backed up to ensure safety. For example, shutdown cooling cannot be maintained just by an erratic isolation condenser system in natural circulation mode as backup power is required for shutdown cooling pumps. Wonder what the response of the reactor to a station blackout is? For sure the story is that the isolation condenser pools are good for 7 days (as required by NRC) and that takes care of all what reactor needs.

Failures and risk assessment of IC systems is subject of much PRA work and many contributors to IC system failures are identified, (see reference xx for example) but no nuclear reactor is ever licensed without provisions of a backup power like emergency diesel generators.

1.15 IRRESPONSIBLE CONCEPT OF "BREAK EXCLUSION ZONE"

A convenient self-assessment of reliability of pairs of isolation valves and an assumption that the valve closure is faster that the speed with which a pie will not destroy them has enabled the vendor to come up with a bizarre (to people who analyze failures) position that almost all big pipes that emanate from the reactor vessel just cannot fail and have created BREAK EXCLUSION ZONES – something that is a good sales pitch but not a responsible position for ensuring that systems and measures are in place to mitigate the effect of pipe and valve failures due to one of dozens of reasons that have historically contributed to those failures in process industries. An example is illustrated in

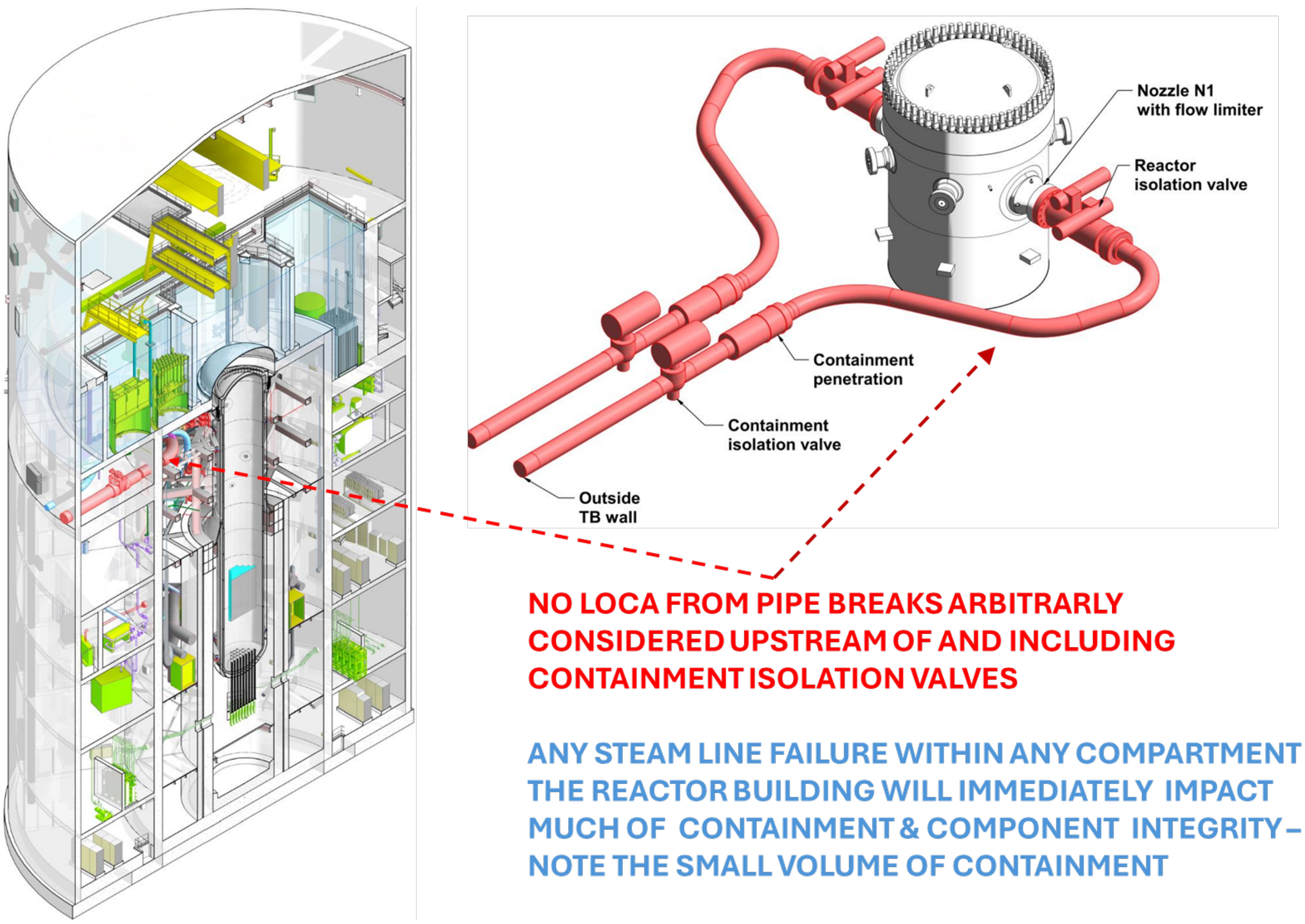


Figure 6 : Example of one of many 'Break Exclusion zones' claimed by GE Hitachi.

Safety assessments of nuclear reactors have for over 60 years of power reactor operation always included the effects of Loss of Coolant Accidents (LOCAs) of various magnitudes to assess the response of reactor

Safety assessments of nuclear reactors have for over 60 years of power reactor operation always included the effects of Loss of Coolant Accidents (LOCAs) of various magnitudes to assess the response of reactor fuel, process and safety systems including the containment and to demonstrate compliance of component and system designs with a multitude of codes and standards developed literally over a century of thoughtful scientific and engineering work. This is done using detailed numerical methods and design assessments, testing, simulations and confirmatory evidence, and not just wishful thinking years before a detailed design covering all well-established norms and standards by a nuclear reactor design shop supervised by a competent regulatory staff. Unfortunately for us that process did not exist. The revolving door between the regulatory staff and the industry, hit common sense in the face and shut out public interests.

GE Hitachi (GEH) has created unilaterally and in the history of reactor designs exclusively and hitherto unheard of a **Break Exclusion Zone** whose net effect is that a LOCA from the reactor vessel pressure boundary or the associated piping cannot affect reactor cooling or inject energy into containment. Just like after-life, one wishes it was true or even remotely possible, but as a nation that will suffer the consequences alone (as Japan did after Fukushima). Many professionals that have participated in acceptance of this erroneous claim are doing a disservice to our country and violating many a norm.

The claim is as follows:

All pipes coming in or out of the reactor vessel have isolation valves at their ends beyond the containment and these valves will close upon a loss of coolant signal. A dynamic analysis including energy discharge at initiation of a rupture or a break, effect of pipe whip on damaging the valves themselves and immediate response of the reactor structures to create a cascading failure is ignored. That is so irresponsible.

This is a great sales brochure feature for the compliant but defies common engineering sense and its blanket acceptance without analysis by CNSC staff defies their legislated duties to be mindful of risk to public, given the trail of failures in reactors designed and built previously.

CNSC staff have effectively agreed that only portions of piping beyond and including the outboard Containment Isolation Valves (CIVs) will be evaluated for postulated breaks and cracks. This effectively removes any consideration of containment and reactor response to Loss of Coolant Accidents (LOCAs) from within design basis accidents and moves it to Beyond Design basis accidents and reduces the containment volume from a needed ~30,000 m^3 to a mere 5,600 m^3.

Imagine giving them a construction license now and then figuring out later hat a much bigger containment is needed. It is prudent to require the vendor to go back to the drawing board and design us a safe reactor using all norms and demonstrating compliance with applicable codes and standards.

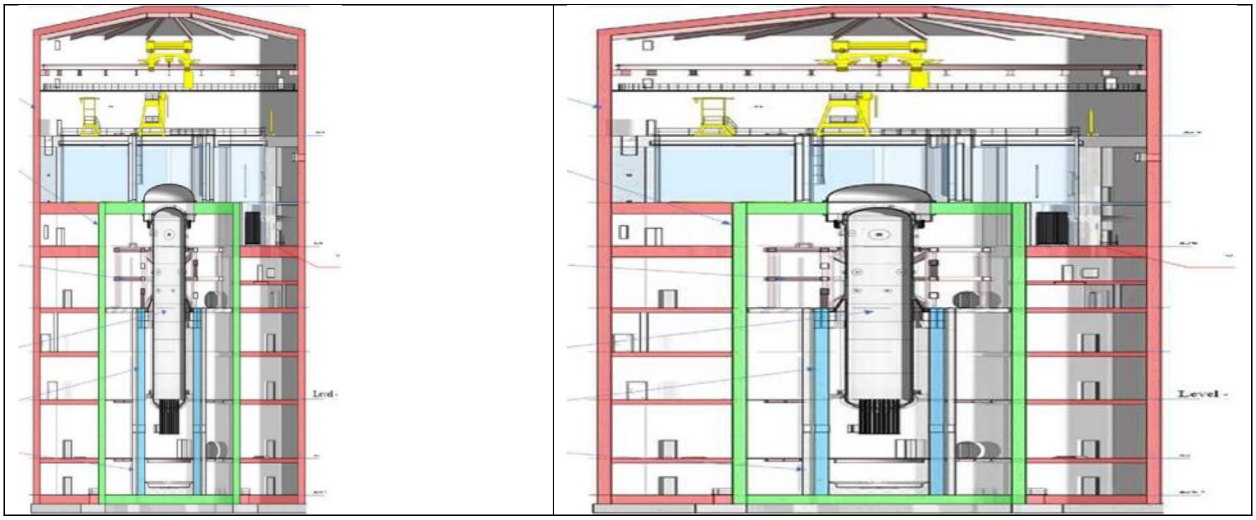


Figure 7 : If a construction starts mow for a reactor building on the left, how will we change it to a larger building on the right?

Thus a medium sized nuclear reactor with reactor power greater than over 100 nuclear reactors that have ever operated and 22 that are still operating (see data and pictures in Appendix xx) without any claims of being 'small', can now claim to be small because the largest single obstruction to post accident energy release – the containment – has been marginalized, shrunken to a volume smaller than that of a municipal water tank.

History is full of claims of engineering perfection that resulted in huge tragedies. My country cannot afford one just because the CNSC staff refuses to do its job and agrees to support a construction license application without a detailed design that can be verified against applicable codes and standards using actual data - not just power point slides or claims from a highly redacted 'safety assessment' to remove 'proprietary' information. This claim of all flange bolted-on Reactor Pressure Vessel (RPV) isolation valves being 'Integral' to the RPV onto which it is bolted makes no engineering sense and has no equivalence in nuclear or chemical or process applications or their risk assessments. The RPV and the RPV-Isolation Valve are 2 different design elements and thus not legal applicants to being infallible as a single-unit entity, ignores the following modes of valve failures.

1.16 TOO MUCH DEPENDENCE ON ISOLATION CONDENSER

Isolation condenser is a critical part of the safety story for the vendors of BWRX-300. It is in contact with the reactor vessel all the time and is likely to be filled with non-condensable gases at the outset. Following schematics in Figure 8 will help the Commissioners understand how it works.

Once the liquid line isolation valves are opened and the contents purged into the reactor vault, a unidirectional flow of "steam in – liquid out" is expected to be established until a time that the water in the pool is low.

A similar "isolation condenser system" did not work as designed in Fukushima, contributing to the triple meltdown in 2011. An early version of an isolation condenser system failed to prevent severe fuel damage in the case of the Humboldt Bay reactor, a BWR with a power of 65MWe, that experienced a serious reactor accident in 1970 triggered by a sudden loss f AC power.

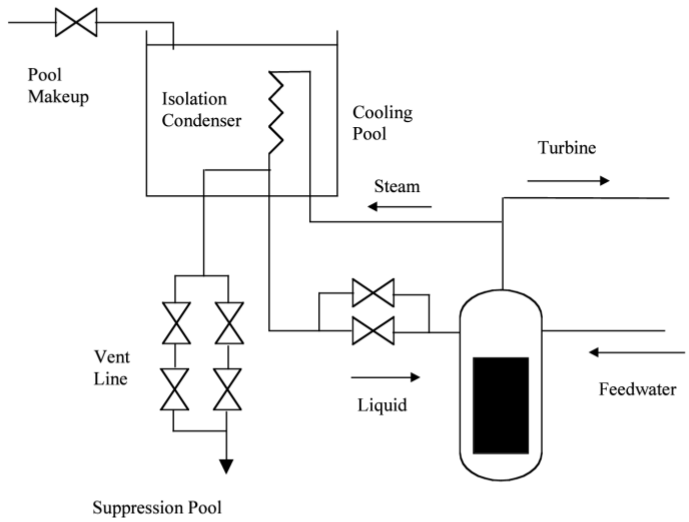


Figure 8 : A Simplified Isolation Condenser system in a normal BWR

It is an old concept of passive heat removal from a steam laden source to a cold heat sink. and has its own risk profile. I have personally done experiments 40 years ago to demonstrate that a phenomenon called Reflux Condensation and effects of accumulated and trapped non-condensable gases (including that from the H_2 addition to the reactor vessel) and counter current flows that can severely inhibit an IC effectiveness.

Figure 10 is a fault tree for a typical IC system. Reference 13 is a good starting point for Commissioners to learn why the IC system can never be allowed to be the only safety system in a reactor that would operate in Canada. Following are some of the factors that affect its failure to perform fully:

1. Failures of actuation devices - components that change state.
2. Failures that defeat or degrade the heat and mass transfer mechanisms.

Even if designed perfectly by divine intervention – such the IC system would still have to meet the design requirements of all applicable codes and regulatory guides to which lesson learnt from BWR accidents at Humboldt Bay and Fukushima, along with those from operating experience from the other reactors must be included. For sure the IC system at Fukushima failed to deliver the promised safety net. We can present more information on that at hearings.

It would be instructive to get data for an 'actual' isolation condenser to see by our own modelling if 6 of them would be sufficient to remove decay heat of 3.7%. We have modelled such heat transfer units before, and the heat transfer area seems to be small for a number of cases where the primary steam pressure may be low.

Figure 9 : A GE Hitachi IC condenser claimed to be sufficient for the purpose.

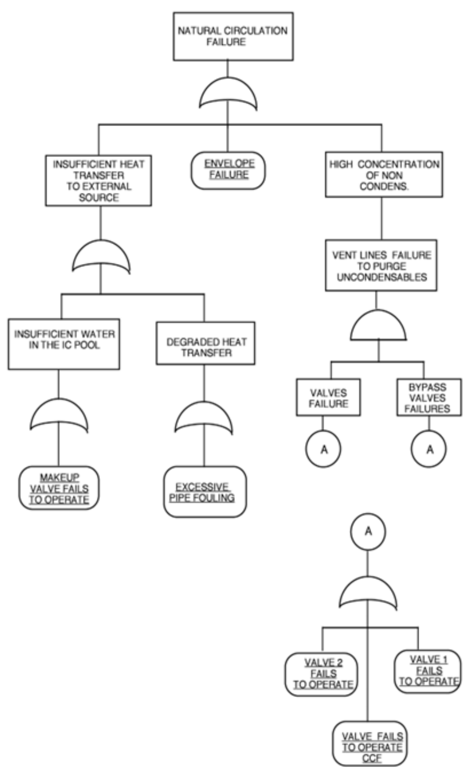


Figure 10 : A sample of Isolation Condenser fault tree depicting failures and contributing factors to a deficient response of Isolation Condenser from reference 13 .

PARTIAL LIST OF QUESTIONS FOR HEARINGS IN JANUARY 2025

2.1 Is a construction permit not premature and against the public interest?

Please have the CNSC staff, that inexplicably recommends issuance of a construction license based on what amounts to be a conceptual design with a lot of hype, without any detailed design, any technical reviews, code compliances and safety assessments as well evaluation of all risks and costs, explain why its decision to do so is not incompatible with its legislated duties to protect public interests by acting in a professional manner and a thoughtful public deliberation of risks and advantages.

There is no information on long term storage plans for spent nuclear fuel assemblies. Feels like being asked to fly a plane whose landing gear hasn't been invented yet.

2.2 Is OPG application not contrary to the US NRC position of waiting?

Please explain why the decision to hold these hearings on a premature issuance of a construction license not incompatible with the clearly stated position of US NRC (with whom it has an agreement to jointly assess the GEH design of BWRX-300) that has required that the BWRX design details and the licensing basis of all is systems and subsystems be sufficiently developed to warrant consideration by NRC staff of any application under CFR-50 and CFR-52 and that even their pre application interactions were far from comprehensive.

2.3 What features are in common with Humboldt Bay and Fukushima reactors??

Please explain why the common features between the design features of BWRX-300 and the core damage accidents in 2 GE designed commercial BWRs (Humboldt Bay 1970 and Fukushima 2011) do not make the BWRX-300 design also vulnerable to similar risks. In both cases, the Isolation Condenser System failed. In the Humboldt case, the ICS failed because a valve failed to open. In the case of Fukushima unit 1, the ICS apparently worked for a while after the earthquake but failed to work after the tsunami.

2.4 Is not the current siting decision contrary to our current understanding of risk?

Please explain why siting of a new reactor type within the already far too small exclusion boundary of Darlington station is not contrary to established safety principles, world practices, lessons learnt from Fukushima and Chernobyl and US CFR 50.100.

2.5 Are the estimates of decommissioning costs at all realistic?

With a legacy of significant core damage accident in the very first commercial BWR at Humboldt Bay in 1970 after only 6 years of not a stellar operating history (cleanup cost to owner utility PG&E$1.6 billion) and multiple BWR reactors at Fukushima (cleanup cost estimates > $500 billion), what are the decommissioning costs?

2.6 Is the reactor design not too incomplete to contermplate a construction licence?

Would you agree that as presented to us, the CNSC and NRC, design of BWRX-300 as a viable nuclear power reactor is largely incomplete and detailed design of major components like the reactor core, reactor vessel, steam separators, containment, IC heat exchangers, piping, valves, heat sinks seems to have not been even conceptualized fully, let alone designed for function, safety, code compliance, longevity and risk reduction?

2.7 Are the claims of low probability for an inside-containment LOCA justifiable?

Please explain why the GEH position on its Break Exclusion Zone for BWRX has any technical merit and how such an unprecedented claim of impossibility of a large LOCA in the containment, not contrary to public interests.

2.8 Why is the BWRX-300 described as a 'small' reactor?

Can CNSC explain to the Canadian public that BWRX-300 at 870 MWth is a small nuclear reactor?

2.9 Is it safe for construction workers to be next to several operating reactors?

Can CNSC explain to the Canadian public that no information exists on measures that need to be taken safety and wellbeing of the ~2000 workers who will work on it for at least 5 years; mere a couple hundred meters from 4 reactor units of designs created about 50 years ago, whose own credible risk assessments for core damage accidents are yet to be undertaken (see appendix xx)? Would CNSC be willing to transfer their office to the construction site?

2.10 What are the plans for procurement of fuel we cannot manufacture??

Can CNSC explain, to safeguard future Canadian public interests what the contingency plans have been considered for contingent fuel procurement in case of disruption to the public utility OPG of disruption of the fuel supply chain? A high enrichment and a proprietary fuel assembly design may not be an easy task for this country.

2.11 Why is OPG allowed to publish a timeline for licensing decision?

Given that a detailed and certified design of the BWRX-300 may be years away, does CNSC not see considerable risk in start building structures for systems that may have to undergo serious revisions and upgrades prior to its certification by a competent regulatory body like the US NRC?

2.12 Should there not be an environmental assessment after the design is complete?

Safeguarding public interest must also include economic interests of the province and the nation. Are the proposed construction activities not subject to environmental assessments? For sure, does CNSC not agree that construction of a reactor located 100 feet below the sub-surface water level must have a different environmental impact than one that is mostly above grade?

2.13 Should there not be a risk assessment and a comprehensive safety analysis?

The safety report made available to us with little information. From what we see that work is incomplete as accidents that typically challenge the reactor safety systems (of which there are so few) are not considered.

It is such a shame that qualified personnel were not assigned by the CNSC brass to this project. Maybe they have all retired and all you have left are people who have never performed a design assessment and have seen independent safety analyses to license a reactor.

2.14 Should there not be a risk assessment for severe accidents?

Looks like there is no consideration of severe accidents and the universal requirement of including lessons learnt from Fukushima are again lost on CNSC staff. I will be happy to brief the Commissioners on the risk that has exposed our public to.

2.15 Should there not be a risk assessment and a comprehensive safety analysis?

Does CNSC staff, especially the professional engineers, understand that their professional designations will be subject to challenge in front of the professional bodies case of incompetence or lying and be individually liable for their conduct in this matter (refer to the case of KEPCO executives held initially liable)?

2.16 How to asses the competence of GE Hitachi/OPG/Atkins-Realis to build the first BWRX?

Our understanding the present day workers in these companies have never built a nuclear reactor. Atkin Realis has the paper rights to CANDUs but no hands on construction experience. Only OPG has some experience and their stellar performance in refurbishment of an old design is commendable.

2.17 What are the lessons learned from Humboldt Bay, Fukushima, Chernobyl, TMI, etc?

A comprehensive design review is handicapped by the following factors:

1. Design of reactor components and systems is incomplete.
2. Very basic information about the design of the reactor is unavailable or withheld.
3. Basic design data, freely available for other SMRs, is treated as top secret.

For example, the Break Exclusion Zone is arbitrarily defined by assuming an infallible union between the pressure vessel and the 4 isolation valves, ignoring totally the more than a dozen different factors that regularly lead to failures of valves attached to pressure vessels, especially those that are subject to operational conditions, such as fluid flows.

The US NRC has clearly stated in multiple assessment reports that their interactions with GEH are only preliminary assessments, and the meeting summaries in Annex B and Annex C suggest the CNSC staff have always seemingly been largely silent participants in these meetings. More detailed assessments by NRC await a completed design with code compliance and more comprehensive safety reports.

2.18 Why is CNSC making the retrieval of information so difficult?

Please explain why the input into the Commission decision-making by members of the general public, including indigenous people has been made so difficult by the CNSC – withholding from them the preliminary design information, correspondence and reports that are otherwise available for all other reactor designs in almost all other jurisdictions around the world. For example, there practically is nothing in these design parameters of BWRX-300 that can be proprietary as there is nothing in conceptual design that is novel or actually designed much beyond the display of 2D computer representations.

REFERENCES

1 US CFR 50.100

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Annex A

Introduction to the common failure modes of valves

What are the most common reasons for valve failure in pressure vessels?

Introduction to the common failure modes of valves

adapted from https://www.redriver.team/common-failure-modes-of-valves/

Valves, which are essential components for controlling the flow of fluids in a wide range of systems, can experience various failure modes that may lead to inefficiencies, leakage, or even system failure. Understanding these potential failures is crucial for ensuring the safety, reliability, and efficiency of operations in industries such as oil and gas, chemical processing, water treatment, and more. Below are some of the most common failure modes of valves and their causes and effects.

Leakage:

Leakage is one of the most common failure modes of valves and can occur both externally and internally.

External Leakage: This type of leakage refers to the escape of fluid from outside the valve, typically from areas such as the stem seal, body gasket, or other external components. External leakage can lead to loss of fluid, inefficiency, and potential hazards, especially if the fluid is toxic or flammable.

Cause: External leakage is often caused by wear or damage to the stem seal or body gasket, improper assembly, or excessive pressure that exceeds the valve's design limits.

Effect: Fluid escapes outside the valve, leading to loss of system pressure, environmental contamination, or safety risks.

Internal Leakage: Internal leakage happens when fluid seeps through a closed valve, which is supposed to stop the flow completely. This can occur due to degraded or damaged seating surfaces or poor valve sealing.

Cause: Internal leakage can be caused by damage to the valve seat or other sealing surfaces, such as wear and tear over time, corrosion, or contamination.

Effect: Loss of control over fluid flow, leading to reduced system performance, wasted energy, or even equipment damage if the leakage goes unnoticed.

Sticking or Binding:

Valves may experience sticking or binding, where they become difficult to operate or even completely immobile. This can affect the valve's ability to control flow properly.

Cause: Sticking or binding usually occurs due to contamination, corrosion, or improper lubrication. These factors can cause friction between moving parts, making the valve difficult to turn or operate.

Effect: The valve becomes difficult to operate, or it may become entirely immobile. In some cases, this can lead to failure in controlling the flow of fluid or gas, causing system inefficiency or potential hazards, especially in critical applications.

Erosion:

Erosion refers to the gradual wear and tear of valve components due to high-velocity flow or the presence of abrasive particles in the fluid passing through the valve.

Cause: High-velocity flow, particularly in systems dealing with abrasive fluids or slurries, can cause rapid degradation of valve components. Abrasive particles strike the valve surfaces, causing mechanical wear.

Effect: Over time, erosion can cause significant damage to the valve’s seating surfaces and internal components, leading to leaks or valve malfunction. Erosion can alsocause the valve to lose its ability to maintain a proper seal, thus affecting the flow control capabilities.

Cavitation:

Cavitation occurs when rapid changes in fluid pressure lead to the formation and collapse of vapor bubbles within the fluid. These vapor bubbles can have damaging effects on the valve components.

Cause: Cavitation typically happens when the pressure of the fluid drops below its vapor pressure, causing bubbles to form. As the pressure increases, these bubbles collapse, creating small but powerful shockwaves that can damage the valve.

Effect: Cavitation can cause pitting or surface damage to valve components, particularly in high-velocity areas. This can weaken the valve, leading to leaks, loss of control, or complete valve failure if the damage is severe.

Corrosion:

Corrosion is a chemical reaction between the valve material and the fluid or the external environment. Corrosion can gradually weaken and degrade valve materials, compromising their performance and longevity.

Cause: Corrosion typically results from the interaction between the valve material (such as steel or iron) and corrosive substances in the fluid or environment. Exposure to chemicals, salts, or moisture can accelerate the corrosion process.

Effect: Corrosion weakens the valve structure, leading to material degradation, leaks, and potential failure. In critical applications, corrosion can be especially dangerous, as it can cause catastrophic failure if left untreated.

Stem Failure:

Stem failure occurs when the valve stem, which connects the valve handle or actuator to the internal mechanisms that control flow, becomes damaged or breaks.

Cause: Stem failure can be caused by excessive torque, misalignment, or corrosion. Over-torquing the valve can place undue stress on the stem, while poor maintenance or improper installation can lead to misalignment.

Effect: When the stem breaks or becomes misaligned, the valve loses its ability to control the flow of fluid. This can lead to safety risks, operational failures, or uncontrolled fluid flow, depending on the application.

Packing Failure:

Packing failure involves the degradation or failure of the valve packing, which is designed to create a seal around the valve stem to prevent leakage.

Cause: Incorrect packing material, overtightening, or normal wear and tear can cause packing failure. The packing material may degrade over time due to exposure to high temperatures, chemicals, or mechanical stress.

Effect: External leakage around the valve stem. When packing fails, fluid can escape around the stem, leading to external leakage. This can reduce system efficiency and pose safety risks, especially in high-pressure or hazardous fluid systems.

Actuator Failure:

Actuator failure is a significant issue in automated valves, where actuators are responsible for opening, closing, or modulating the valve based on the system’s requirements.

Cause: Actuator failure can be caused by electrical, pneumatic, or hydraulic issues in automated valves. Electrical faults such as shorts, power supply issues, or control signal errors can prevent the actuator from receiving the necessary input. In pneumatic or hydraulic systems, leaks, pressure imbalances, or mechanical issues can prevent the actuator from functioning correctly. Automated valves rely heavily on actuators to operate correctly, and any disruption to the actuator's power or control system can lead to a complete failure of valve operation. Proper maintenance and regular inspections are critical to avoiding actuator failure.

Effect: Inability to open, close, or modulate the valve. When actuators fail, valves cannot perform their intended function, leading to an inability to control fluid flow. This can cause operational inefficiencies, safety risks, and unplanned downtime as the valve remains stuck in its current position.

Thermal Expansion or Contraction:

Thermal expansion or contraction is a common failure mode that affects valves operating in environments with fluctuating temperatures.

Cause: Rapid temperature changes in the system can cause materials to expand or contract. As the valve's materials heat up, they expand, and when cooled, they contract. Over time, this repeated cycle of expansion and contraction can lead to misalignment of valve components, especially in systems not designed to accommodate temperature variations. Valves exposed to extreme heat or cold are particularly susceptible to this failure mode. If the materials used in valve construction cannot withstand these temperature changes, they may warp, degrade, or fail.

Effect: Misalignment or binding of valve components. When thermal expansion or contraction occurs, it can cause parts of the valve to become misaligned, leading to difficulties in opening or closing the valve or even complete valve seizure.

Galling:

Galling occurs when two metal surfaces rub against each other without adequate lubrication, causing surface damage.

Cause: Galling is caused by the sliding contact between metallic surfaces without proper lubrication. This often happens in valve components such as the stem and body,

where movement is required to open or close the valve. Without adequate lubrication, the friction between these metal parts increases, leading to wear and surface damage. Galling is particularly common in stainless steel valves and can occur when materials with similar hardness levels are used. The lack of lubrication allows metal to transfer between surfaces, causing the surfaces to seize or become rough.

Effect: Wear and potential seizing of valve components, such as between the stem and the body. This can lead to difficulties in valve operation, increased wear on other components, and ultimately, valve failure if not addressed.

Over-pressurization:

Over-pressurization occurs when the pressure inside the valve exceeds the valve's rated capacity, potentially leading to catastrophic failure.

Cause: Over-pressurization is caused by exposure to pressure levels beyond the valve's rated capacity. This can occur due to system malfunctions, operator error, or unexpected spikes in pressure. Valves are designed to operate within specific pressure limits, and exceeding these limits can lead to material deformation, cracking, or bursting.

Over-pressurization is particularly dangerous in systems that operate with gases or liquids under high pressure, as the energy stored in these fluids can cause explosive failures if the valve fails.

Effect: Potential deformation or bursting of the valve body. Over-pressurization can cause the valve to leak, crack, or, in extreme cases, explode, leading to severe damage to the system and posing significant safety risks to personnel and equipment.

Elastomer Degradation:

Elastomer components in valves, such as seals and gaskets, are vulnerable to degradation when exposed to harsh chemicals, extreme temperatures, or excessive wear.

Cause: Elastomer degradation occurs when seals or gaskets are exposed to temperatures or chemicals beyond the elastomer's capacity. Over time, elastomers can become brittle, crack, or lose their elasticity, making them less effective at sealing and preventing leaks. Certain chemicals or aggressive media can accelerate the degradation process, breaking down the elastomer's molecular structure and causing it to lose its sealing capabilities. This is especially common in applications involving corrosive substances or extreme temperatures.

Effect: Seal failures leading to internal or external leakage. When elastomers degrade, they can no longer provide a reliable seal, resulting in leaks that compromise the system's integrity and efficiency. In some cases, the failure of elastomer components can lead to catastrophic system failure.

Seat Wear or Damage:

The valve seat is a critical component that provides the sealing surface for the valve. Wear or damage to the seat can lead to leakage and operational issues.

Cause: Regular operation, debris in the fluid, or cavitation can cause wear or damage to the valve seat. As the valve opens and closes repeatedly, the seat can become worn down, particularly in high-pressure systems or systems with abrasive fluids. In addition to regular wear, cavitation can cause pitting or surface damage to the seat, further reducing its effectiveness as a sealing surface.

Effect: Compromised sealing capability, leading to internal leakage. As the valve seat becomes worn or damaged, the valve may no longer seal properly, allowing fluid to leak through even when the valve is supposed to be closed.

Improper Installation:

Improper installation can lead to a variety of valve failures, from misalignment to leakage or reduced efficiency.

Cause: Improper installation can result from misalignment, incorrect gasket selection, or incorrect installation practices. When a valve is not installed according to the manufacturer’s specifications, it may not function as intended, leading to premature failure. Incorrect gasket selection can cause leakage, while misalignment can result in excessive wear on valve components or difficulty in operating the valve.

Effect: Reduced valve efficiency, increased wear, or leakage. Improper installation can lead to various operational issues, including difficulty in opening or closing the valve, reduced system performance, or even catastrophic valve failure if critical components are installed incorrectly.

Material Incompatibility:

Valves must be made from materials that are compatible with the media they will be handling to prevent corrosion, erosion, and failure.

Cause: Material incompatibility occurs when pressure vessel manufacturers fail to adequately consider the media or environment in which the valve operates. For instance, if a valve designed for water service is exposed to acidic or basic solutions, the materials may not hold up against the harsh conditions, leading to accelerated corrosion, erosion, or wear. Incompatible materials can degrade quickly in certain environments, leading to reduced valve lifespan and frequent maintenance or replacement needs.

Effect: This can lead to accelerated corrosion, erosion, and wear. For instance, a valve designed for water service might fail prematurely if exposed to acidic or basic solutions. Material incompatibility can result in frequent leaks, reduced performance, and potential system contamination if the valve materials react with the fluid.

Design Flaws:

Design flaws in the valve can result in operational issues, inefficiency, or even catastrophic failure.

Cause: Design flaws can occur when pressure vessel manufacturers overlook specific application requirements, leading to inadequacies in the valve's design. For instance, the valve may not be designed to handle the system's pressure or temperature requirements, or it may not be sized correctly for the flow rate. Poor design can also result in turbulence, premature wear, or improper sealing, reducing the valve's efficiency and lifespan.

Effect: Such oversights can result in issues like turbulence, premature wear, or even catastrophic failures if the valve can't handle the system's pressures. Valves with

design flaws may need to be replaced sooner than expected, leading to increased downtime and maintenance costs.

Inadequate Testing:

Cause: Insufficient quality control by the pressure vessel manufacturer during the production phase.

Effect: Valves might have hidden defects or weaknesses that only manifest under operational conditions, leading to unexpected system downtimes or safety concerns.

Maintenance Overlook:

Cause: Neglecting regular inspections and maintenance, sometimes due to a false sense of security provided by high standards set by pressure vessel manufacturers.

Effect: Wear and tear go unnoticed, and the likelihood of failure increases.

Improper Sizing:

Cause: A mismatch between the valve's size and the system's requirements, which can arise if a pressure vessel manufacturer doesn't provide clear specifications or if the end-user misinterprets them.

Effect: Reduced flow efficiency, excessive pressure drops, or even valve damage due to over-pressurization.

Faulty Actuation Mechanisms:

Cause: Defective or inadequate actuators provided by the pressure vessel manufacturer.

Effect: Automated valves may not operate as required, leading to control issues or safety hazards.

Environmental Factors:

Cause: External conditions like high humidity, saline environments, or temperature extremes, which might not be factored in by the pressure vessel manufacturer.

Effect: Accelerated corrosion, elastomer degradation, or mechanical failures due to material contraction/expansion.

Operational Errors:

Cause: Human errors in handling or operating the valve, especially if instructions from the pressure vessel manufacturer are unclear or not followed.

Effect: Accidental over-torquing, improper cycling, or other mishandlings can damage the valve or compromise its performance.

In the realm of pressure vessels, the quality, reliability, and durability of valves are paramount. Faulty valves can compromise the integrity of the entire pressure system, posing potential safety risks. Hence, pressure vessel manufacturers must place significant emphasis on the design, production, and testing of valves. Additionally, end-users and maintenance teams should be well-informed and trained about proper installation, operation, and upkeep practices. Collaborative efforts between pressure vessel manufacturers and users can ensure that valves function optimally, thereby safeguarding the longevity and safety of the entire system.

What are the most common reasons for valve failure in pressure vessels?

Valve failure in pressure vessels often occurs due to factors like corrosion, improper installation, material fatigue, and operational errors. Corrosion can weaken valve components, leading to leaks or breaks. Improper installation might result in misalignment or undue stress on valve parts. Material fatigue happens over time due to repeated stress, while operational errors can include incorrect handling or exceeding operational limits of the valves.

How does corrosion impact valve performance in pressure vessels?

Corrosion is a significant threat to valve integrity in pressure vessels. It can lead to the deterioration of metal parts, causing leaks or blockages in the valve mechanism. Corrosion typically occurs due to chemical reactions between the valve material and the substances inside the pressure vessel or environmental factors. Regular inspection and using corrosion-resistant materials are key to mitigating this risk.

Can improper installation lead to valve failure?

Yes, improper installation is a critical factor in valve failure. If a valve is not aligned correctly or is installed without proper seals, it can lead to leaks or pressure imbalances. This misalignment can also cause undue stress on certain parts of the valve, accelerating wear and tear. Ensuring that valves are installed by experienced technicians following manufacturer guidelines is essential for their longevity.

What role does material selection play in preventing valve failure?

Material selection is crucial in preventing valve failure. The materials used for valves must be compatible with the contents of the pressure vessel and the operating environment. For instance, valves in vessels containing corrosive substances should be made of corrosion-resistant materials. Similarly, high-pressure applications require materials that can withstand significant stress without deforming or breaking.

How can operational errors lead to valve failure in pressure vessels?

Operational errors, such as exceeding the designed pressure limits, rapid cycling, or incorrect handling, can lead to valve failure. Exceeding pressure limits can cause stress beyond what the valve is designed to handle, leading to material failure. Rapid cycling can result in excessive wear and tear, while incorrect handling might involve using the wrong valve type for specific applications, leading to malfunction.

Annex B

NRC Summary of Meeting, January 24, 2024

BEH Presentation: BWRX-300 Break Exclusion Zone

From: Jim Shea
Sent: Thursday, February 8, 2024 6:50 AM
To: GEH-BWRX-300MtgSumPEm Resource
Cc: Jordan Glisan
Subject: January 24, 2024 Meeting Summary regarding the GEH approach to the BEZ for the BWRX-300
Attachments: January 24, 2024 Public Closed Meeting Summary on NRC Staff Feedback on GEH BEZ for BWRX-300.docx; BTP 3-3 & 3-4 Public Meeting Slides.pdf

On January 24, 2024, from 10:00am to 11:30am, the NRC held an open and partially closed meeting with GEH regarding the application of NRC SRP BTP 3-3 & 3-4 to the piping configuration associated with the BWRX-300 SMR design.

The public / partially closed meeting summary is attached along with the GEH presentation and piping configuration that is the subject of the meeting.

If there are any further questions regarding this subject, please contact the NRC project manager.

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U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
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Washington, DC, 20555-0001

Annex B

Hearing Identifier: GEH_BWRX300_Mtgs_Public
Email Number: 29

Mail Envelope Properties (BLAPR09MB6899D4D0E3F4FCC5A764754E94442)

Subject: January 24, 2024 Meeting Summary regarding the GEH approach to the BEZ for the BWRX-300
Sent Date: 2/8/2024 6:50:18 AM
Received Date: 2/8/2024 6:50:25 AM
From: Jim Shea

Created By: James.Shea@nrc.gov

Recipients:
"Jordan Glisan" <Jordan.Glisan@nrc.gov>
Tracking Status: None
"GEH-BWRX-300MtgSumPEm Resource"
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Tracking Status: None

Post Office: BLAPR09MB6899.namprd09.prod.outlook.com

Files	Size	Date & Time
MESSAGE	753	2/8/2024 6:50:25 AM
January 24, 2024 Public Closed Meeting Summary on NRC Staff Feedback on GEH BEZ for BWRX-300.docx	54455	
BTP 3-3 & 3-4 Public Meeting Slides.pdf	1595986	

Options
Priority: Normal
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date:

U.S. Nuclear Regulatory Commission Public Meeting Summary

Title:

Pre-application meeting with GE-Hitachi Nuclear Energy Americas, LLC (GEH) on U.S. Nuclear Regulatory Commission (NRC) staff NUREG 0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP) Branch Technical Positions (BTP) 3-3 & 3-4.

Meeting Notice: Agency Document Accession Management System (ADAMS) Accession No. ML24023A307.

Date of Meeting: Wednesday, January 24, 2024.

Location: Via teleconference Microsoft Teams.

Type of Meeting: Observation / Partially Closed.

Purpose of the Meetings:

The purpose of this pre-application meeting is to have a discussion of GEH's proposed alternative to NRC staff SRP branch technical position BTP 3-3 & 3-4, Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment.

Meeting objectives are to:

- Provide a high-level overview of the BWRX-300 containment penetration area configuration.

- Seek NRC feedback on the approach to the Break Exclusion Zone (BEZ) for the BWRX300.

Summary of Meeting:

On January 24, 2024, an observation public and partially closed meeting was held between the U.S. Nuclear Regulatory Commission (NRC) staff and GEH concerning the application of SRP BTP 3.3 & 3-4, to the GEH BWRX-300 piping configurations. This topic is related to the NRC approved Pre-Application License Topical Report (LTR), NEDO–33910P-A, Revision 2, BWRX-300, "Reactor Pressure Vessel Isolation and Overpressure Protection." The LTR was submitted on December 31, 2019 (ADAMS Accession No. ML20174A577) and was approved by the staff in a letter dated November 18, 2020 (ADAMS Accession No. ML20310A153). The public NRC staff approved -A LTR can be located on the NRC public website under small modular reactor (SMR) Pre-Application Activities GEH BWRX-300 (ADAMS Accession No. ML23167A086). In the LTR, the staff accepted the design functions described for the BWRX-300 SMR in general specifically related to the isolation and pressure relief aspects. In addition, GEH described its BWRX-300 design requirements to identify postulated pipe rupture locations and configurations inside containment as specified in SRP Chapter 3, "Design of Structures, Components, Equipment, and Systems," BTP 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," Revision 3, December 2016, Part B, Item 1(iii)(2), and identifying leakage cracks as specified in BTP 3-4, Part B, Item 1(v)(2). However, GEH stated, “Design details are to be described during future licensing activities and the staff noted that, “Specific aspects of the connection of the RPV isolation valves to the reactor vessel will be reviewed during future licensing activities of the BWRX-300 SMR.” The staff further stated in its safety evaluation report that, “If an applicant for a construction permit under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities,” or a design certification or combined license (COL) under 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” is not able to demonstrate compliance with an NRC regulation when the detailed design of the BWRX-300 SMR is complete, the applicant will be expected to justify an exemption from the applicable regulatory requirement. The NRC staff will evaluate the regulatory compliance of the final design of the RPV isolation and overpressure protection features for the BWRX-300 SMR and the piping rupture locations or break exclusion zone(s) (BEZ) proposed by GEH during future licensing activities in accordance with 10 CFR Part 50 or 10 CFR Part 52, as applicable. As discussed in the approved LTR safety evaluation, GEH indicated in the submittal that the detailed design of the BWRX-300 SMR including the piping configuration inside and outside containment is not complete therefore, the NRC staff will make a final determination of the BWRX-300 SMR’s acceptability when the detailed design is completed and reviewed by the NRC staff during future licensing activities.

The meeting commenced with a brief introduction by the NRC licensing project manager, who explained the purpose of the meeting, provided background on the approved staff LTR NEDO-33910P-A, Revision 2, and briefly introduced the NRC staff participating in the meeting and described the meeting logistics. All GEH staff participated remotely. NRC staff participated in the meeting at the designated conference room and also remotely.

In addition, The NRC introduced the Canadian Nuclear Safety Commission (CNSC) review team participating in this meeting remotely, as part of a 2019 memorandum of cooperation between the NRC and CNSC, and a September 2022, “Charter - Collaboration on GEH’s BWRX-300 Design.” Therefore, as an integral part of the meeting the CNSC can ask and participate in discussions with GEH during the public portion as well as the closed portion of the meeting.

The principal CNSC technical staff members who attended included:

- Mazhar, Hazem hazem.mazhar@cnsc-ccsn.gc.ca
- Eom, Seyun seyun.eom@cnsc-ccsn.gc.ca

GEH provided a public overview of the BWRX-300 reactor pressure vessel (RPV) piping configurations (ADAMS Accession No. ML24016A302) submitted to the NRC in a letter dated January 16, 2024 (ADAMS Accession No. ML24016A301). In its presentation GEH provided some of the piping detail that was not part of the approved LTR NEDO–33910P-A, Revision 2, which was used during the meeting discussions.

GEH specifically was seeking NRC staff feedback on its proposed alternate approach from its approved LTR, for meeting the requirements in 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” (GDC) 4, “Environmental and Dynamic Effects Design Bases,” which requires, in part, that structures, systems, and components (SSCs) important to safety be designed to accommodate the effects of postulated accidents, including appropriate protection against the dynamic effects of postulated pipe ruptures. Guidelines are provided in NRC NUREG-0800 “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition),” Chapter 3, “Design of Structures, Components, Equipment, and Systems, Sections 3.6.1 and 3.6.2 and associated Branch Technical Positions (BTP) 3-3, 3-4. Specifically, BTP 3-4, “Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment,” Revision 3, December 2016.

GDC 4 allows the use of analyses reviewed and approved by the Commission to eliminate from the design basis the dynamic effects of the pipe ruptures postulated in SRP Section 3.6.2, “Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping,” Revision 3, December 2016. In SRP Section 3.6.3, “Leak-Before-Break Evaluation Procedures,” Revision 1 March 2007, it states that, “The staff reviews and approves the plant-specific piping system submitted from licensees and applicants to eliminate these dynamic effects [Break Exclusion Zone (BEZ)]. A staff approved leak-before-break (LBB) analysis permits licensees to remove protective hardware such as pipe whip restraints and jet impingement barriers, redesign pipe connected components, their supports and their internals, and other related changes in operating plants. Likewise, requirements for plants under construction or being designed are similarly relaxed. The staff’s review ensures that adequate consideration has been given to direct and indirect pipe failure mechanisms and other degradation sources which could challenge the integrity of piping. The staff reviews the direct pipe plant specific and configuration specific failure mechanisms and fracture mechanics analyses.

GEH started the meeting discussions with the outline of its planned approach for applying a BEZ to the design configuration for the BWRX-300 that would apply SRP BTP 3-4 Rev.3 Section B.1.(ii) related to Fluid System Piping in Containment Penetration Areas which states that, “Breaks and cracks need not be postulated in those portions of piping from containment wall to and including the inboard or outboard isolation valves, provided that they meet the design criteria of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section III, Subarticle NE-1120, and the additional design criteria listed as well as criteria 4 of BTP 3-4 Section B.1.(ii): “The length of these portions of piping should be reduced to the minimum length practical.” This alternative approach from the approved LTR NEDO–33910P-A, Revision 2, was the primary subject of this meeting.

GEH proposed that, “Consistent with current industry practice, that the BWRX-300 will apply the BWRX-300 BEZ from the inboard containment isolation valve to the nearest seismic anchor beyond the outboard isolation valve. GEH also stated that the BEZ pipe length is minimized and associated routing is optimized to achieve sufficient flexibility and minimize stresses given thermal expansion and anchor motions. The BWRX-300 RPV Isolation Valves (which also serve as containment inboard isolation valves) are located integral to the RPV which GEH states is a design element to restore the reactor coolant boundary integrity in the event of a loss of coolant accident condition. GEH pointed to previous precedents from the NRC staff that evaluated piping segments for BEZ treatment that would bound the GEH BWRX-300 piping lengths expected in the final detailed design of the BWRX-300.

The NRC staff did not have a specific objection to this interpretation of BTP 3-4, however it is noted that approval of the elimination of dynamic effects from postulated pipe ruptures is obtained individually for particular piping systems and particular design configurations. LBB is applicable only to an entire piping system or analyzable portion thereof. LBB cannot be applied to individual welded joints or other discrete locations. Analyzable portions are typically segments located between piping anchor points. When LBB technology is applied, all potential pipe rupture locations are examined. The examination is not limited to those postulated pipe rupture locations determined from SRP Section 3.6.2. LBB analyses should demonstrate that the probability of pipe rupture is extremely low under conditions consistent with the design basis for the piping. A deterministic evaluation of the piping system that demonstrates sufficient margins against failure, including verified design and fabrication and an adequate in-service inspection program, including leakage detection capabilities, can be assumed to satisfy the extremely low probability criterion. Also, according to NRC BTP 3-3, “Protection Against Postulated Piping Failures in Fluid Systems Outside Containment,” it states that, “Even though portions of the main steam and feedwater lines meet the break exclusion requirements of item 2.A(ii) of BTP 3-4, [BEZ] they should be separated from essential equipment. Designers are cautioned to avoid concentrating essential equipment in the break exclusion zone. Essential equipment must be protected from the environmental effects of an assumed nonmechanistic longitudinal break of the main steam and feedwater lines. Each assumed nonmechanistic longitudinal break should have a cross sectional area of at least one square foot and should be postulated to occur at a location that has the greatest effect on essential equipment. In the staff approved LTR NEDO–33910P-A, Revision 2, there was no discussion, and the staff did not make any regulatory findings on the application of BTP 3-3 or BTP 3-4 B.1.(ii) for the BWRX-300. As previously described the LTR did not include sufficient design information for NRC staff to make conclusive findings related to either BTP 3-3, 3-4, or the application of LBB analysis to determine specific BEZ applicability to the BWRX-300 design. In addition, the staff pointed to LBB criteria related to fatigue cracking or failure to ensure that the potential for pipe rupture due to thermal and mechanical induced fatigue is unlikely. Licensees and applicants must demonstrate that (a) adequate mixing of high and low temperature fluids occurs in the piping so that there is no potential for cyclic thermal stresses, and (b) there is no potential for vibration-induced fatigue cracking or failure.

At the conclusion of the open portion of the meeting the public was invited to make comments or ask questions of the NRC staff and since there were no comments or question from members of the public, the NRC staff continued the meeting in a closed session continuing to discuss at more depth the questions raised during the public open portion of the meeting as well as some discussion on the specific isolation valve configuration as depicted in the GEH presentation diagrams.

One of the points that was brought out during the closed session from the NRC staff is that following acceptance of an applicant’s BEZ only dynamic effects of postulated pipe ruptures may be eliminated when LBB technology is shown to be applicable. Requirements for containment design, emergency core cooling system performance, and environmental qualification of electrical and mechanical equipment are not affected. In addition, the staff discussed some details of the isolation valves with GEH and as was stated in the staff safety evaluation report of the LTR, “The NRC staff will review the specific aspects of the connection of the RPV isolation valves to the reactor vessel during future licensing activities of the BWRX-300 SMR.” In addition to the NRC staff discussions with GEH the CNSC had similar and more specific discussions of the regulatory treatment and requirements that are specific to Canada. Since the CNSC was not specifically acquainted with the NRC guidance found in BTP 3-3, 3-4 and the LBB requirements the staff is planning to have a discussion separately with CNSC on these subjects.

There were no regulatory decisions made as a result of this public meeting.

The meeting was then adjourned as scheduled at 11:30 am.

Meeting participants: See the following page.

GE-Hitachi Nuclear Energy

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January 16, 2024

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555-0001

Subject: Presentation Slides for Pre-Application Meeting for GE Hitachi Nuclear Energy BWRX-300 Proposed Break Exclusion Zone Methodology and Application Requirements

Enclosed are the presentation slides for the pre-application meeting to discuss proposed Break Exclusion Zone (BEZ) methodology and application requirements for the GE-Hitachi Nuclear Energy Americas, LLC (GEH) BWRX-300 to be held on January 24, 2024.

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

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

Sincerely,

George E. Wadkins

George E. Wadkins
Chief Consulting Engineer - Licensing
GE Hitachi Nuclear Energy Americas, LLC

Enclosures:

1. Presentation Slides for Pre-Application Meeting for GE Hitachi Nuclear Energy BWRX-300 Proposed Break Exclusion Zone Methodology and Application Requirements – Non-Proprietary Information

cc: J. Shea, US NRC

PLM Specification 008N3519 Revision 0

Document Components:
001 M240010 Cover Letter.pdf
002 M240010 Enclosure 1 Non-Proprietary.pdf

ENCLOSURE 1

M240010

Presentation Slides for Pre-Application Meeting for GE Hitachi Nuclear Energy BWRX-300 Proposed Break Exclusion Zone Design Requirements

Non-Proprietary Information

Non-Proprietary Information

BWRX-300
Break Exclusion Zone (BEZ)

January 24, 2024

Non-Proprietary Information

Objective

- Provide a high level overview of the BWRX-300 containment penetration area configuration
- Seek NRC feedback on the approach to the Break Exclusion Zone (BEZ) for the BWRX-300

Annex B

Non-Proprietary Information

Postulation of Pipe Ruptures

- The postulation of pipe ruptures is specified in NRC regulation and guidance:
  - 10CFR50, Appendix A, General Design Criterion 4, "*Environmental and dynamic effects design bases*"
  - NRC Standard Review Plan (SRP, NUREG-0800):
    - Sections 3.6.1 and 3.6.2 , Rev.3
    - 3.6.3 , Rev.1
    - BTP 3-3, Rev.3
    - BTP 3-4 , Rev.3

Annex B

Non-Proprietary Information

Break Exclusion Zone (BEZ)

BTP 3-4 Rev.3 section B.1.(ii)

Fluid System Piping in Containment Penetration Areas.

"Breaks and cracks need not be postulated in those portions of piping from containment wall to and including the inboard or outboard isolation valves, provided they meet the design criteria of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Subarticle NE-1120, and the following additional design criteria: …"

(1) Reduced stress and fatigue limits.
(2) Avoidance of welded attachments, or detailed stress analysis or tests.
(3) Minimized number of circumferential or longitudinal welds.
(4) Lengths of these portions of piping reduced to the minimum practical.
(5) Pipe anchors, restrains, attachments to penetrations and whip restraints not welded directly to the pipe except where 100% volumetrically examinable and detailed stress analyzed.
(6) Requirements on guard pipes.
(7) A 100 percent volumetric inservice examination of all pipe welds should be conducted during each inspection interval as defined in ASME Code, Section XI, IWA-2400.

Annex B

Break Exclusion Zone (BEZ)

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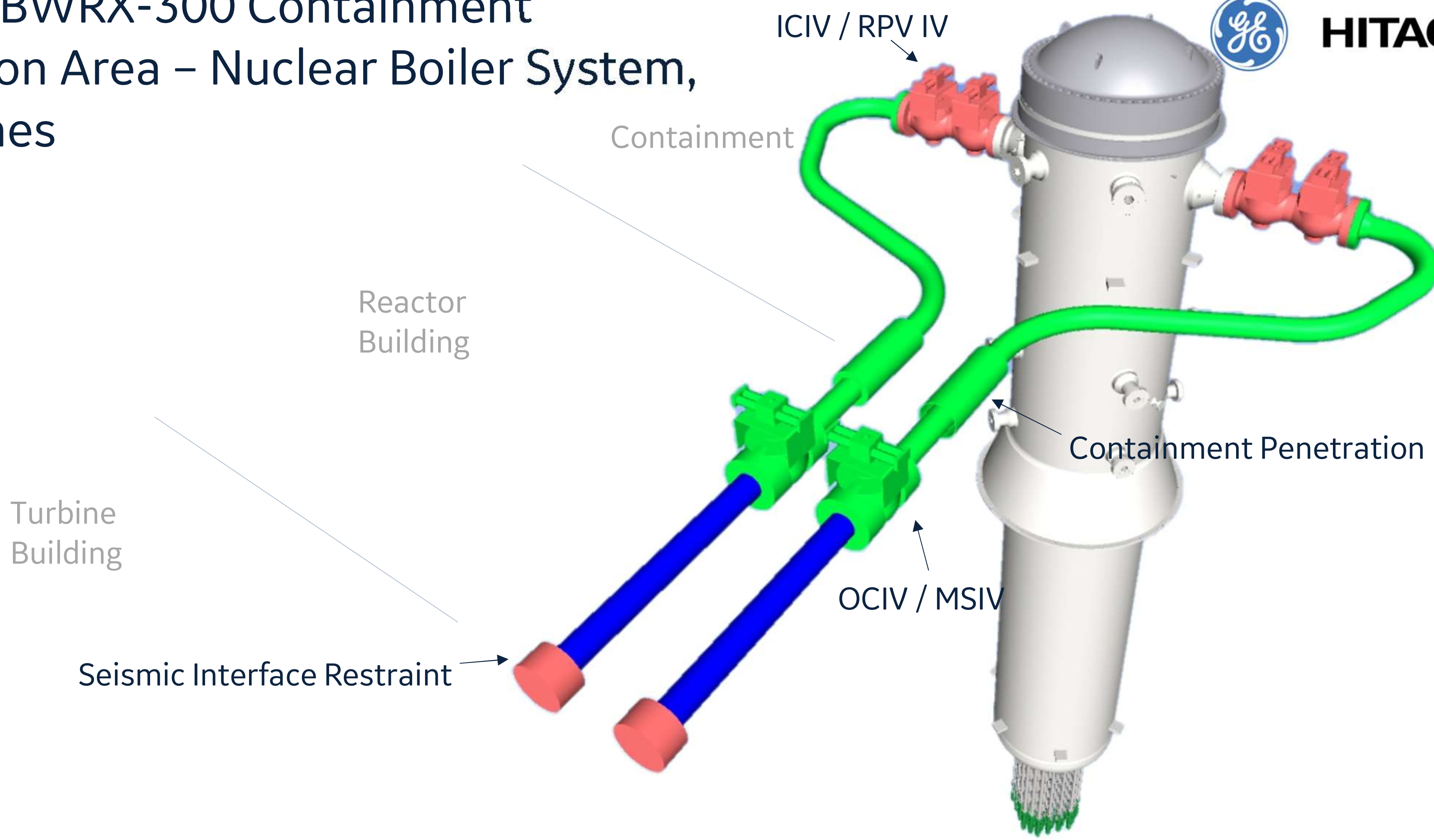
Consistent with current industry practice, the BWRX-300 applies BEZ from the inboard containment isolation valve to the nearest anchor beyond the outboard isolation valve:

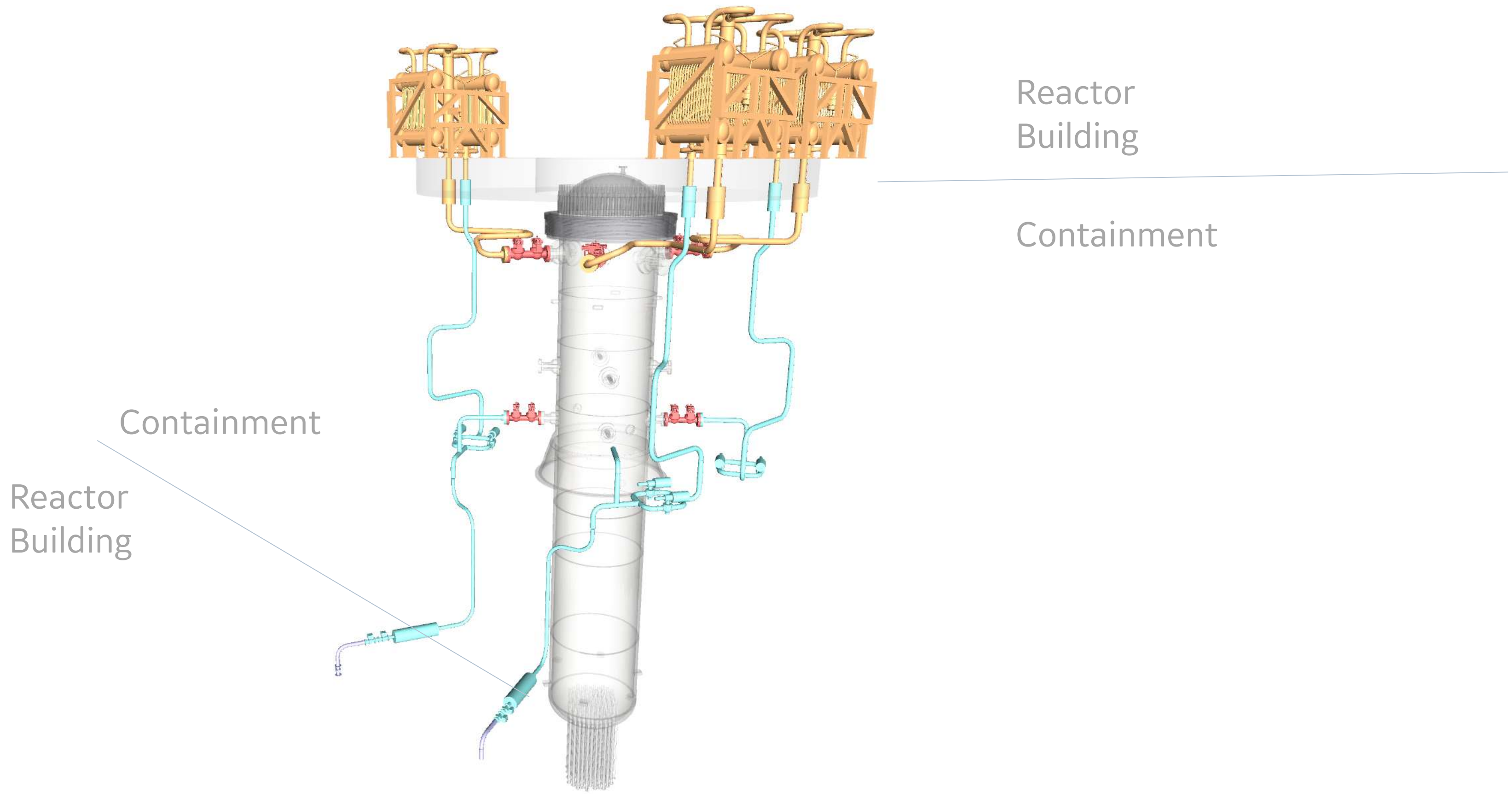
1. The BWRX-300 RPV Isolation Valves (which also serve as containment inboard isolation valves) are located integral to the RPV in order to restore the reactor coolant boundary integrity. This ensures fuel cooling in the case of a loss of coolant accident as opposed to solely relying on the containment boundary in the event of a loss of coolant.
2. The functional boundaries of the BEZ, according to the BTP 3-4, are from inboard (inside containment) isolation valve to outboard (outside containment) isolation valve.
3. Additionally, the structural boundaries of the BEZ are the anchors that extend beyond the outboard isolation valves up to Seismic Interface Restraint (SIR) anchors. These anchors protect the containment isolation valve function from high energy line breaks and seismic effects beyond the SIRs.
4. Accordingly, the piping between the OCIV and the SIR maintains the enhanced piping classification and requirements of the functional boundaries of the BEZ.
5. Pipe length is minimized and associated routing is optimized to achieve sufficient flexibility and minimize stresses given thermal expansion and anchor motions.

Example: BWRX-300 Containment Penetration Area – Nuclear Boiler System, Steam Lines



Non-Proprietary Information

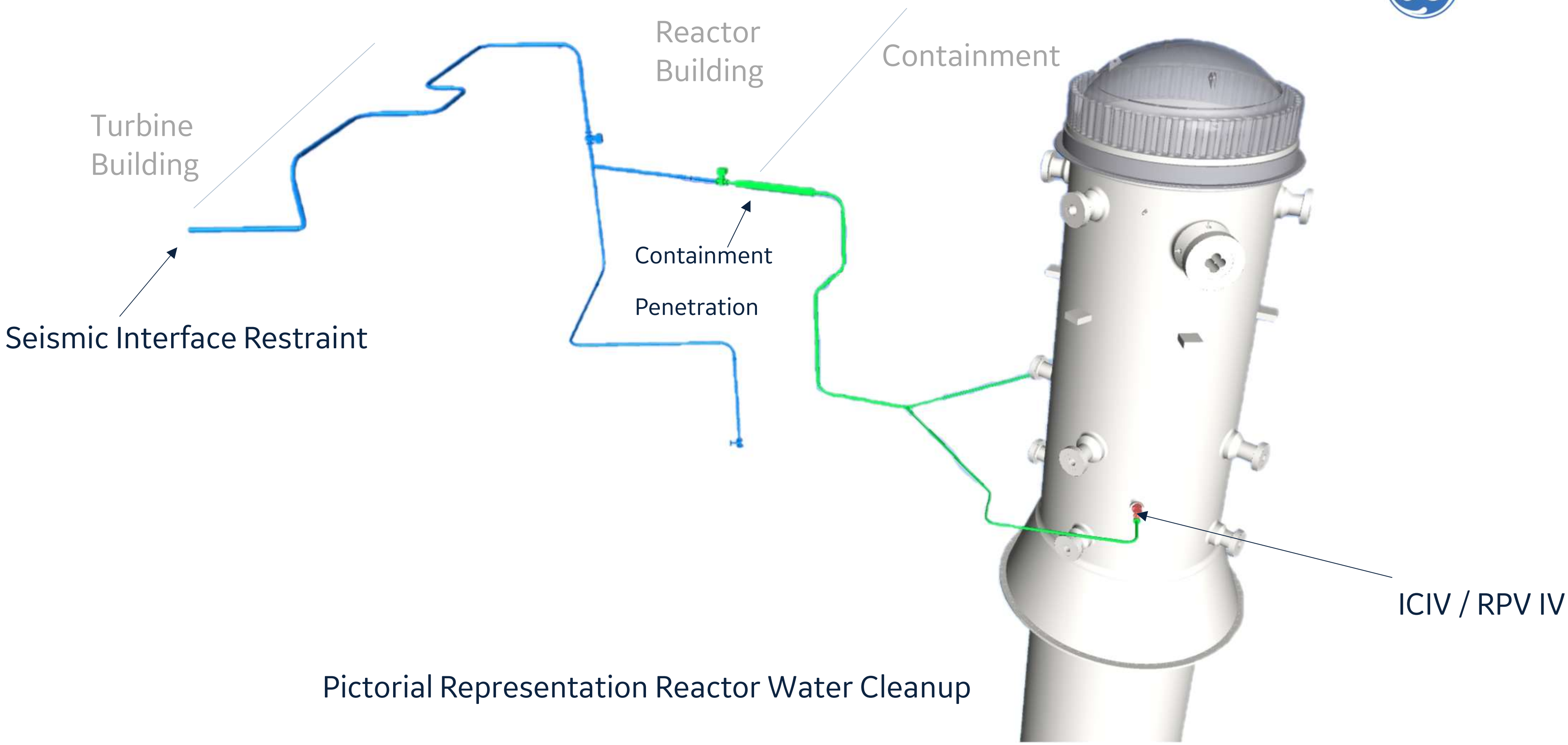
BWRX-300 Isolation Condenser System - BEZ



Pictorial Representation of ICS Line

Non-Proprietary Information

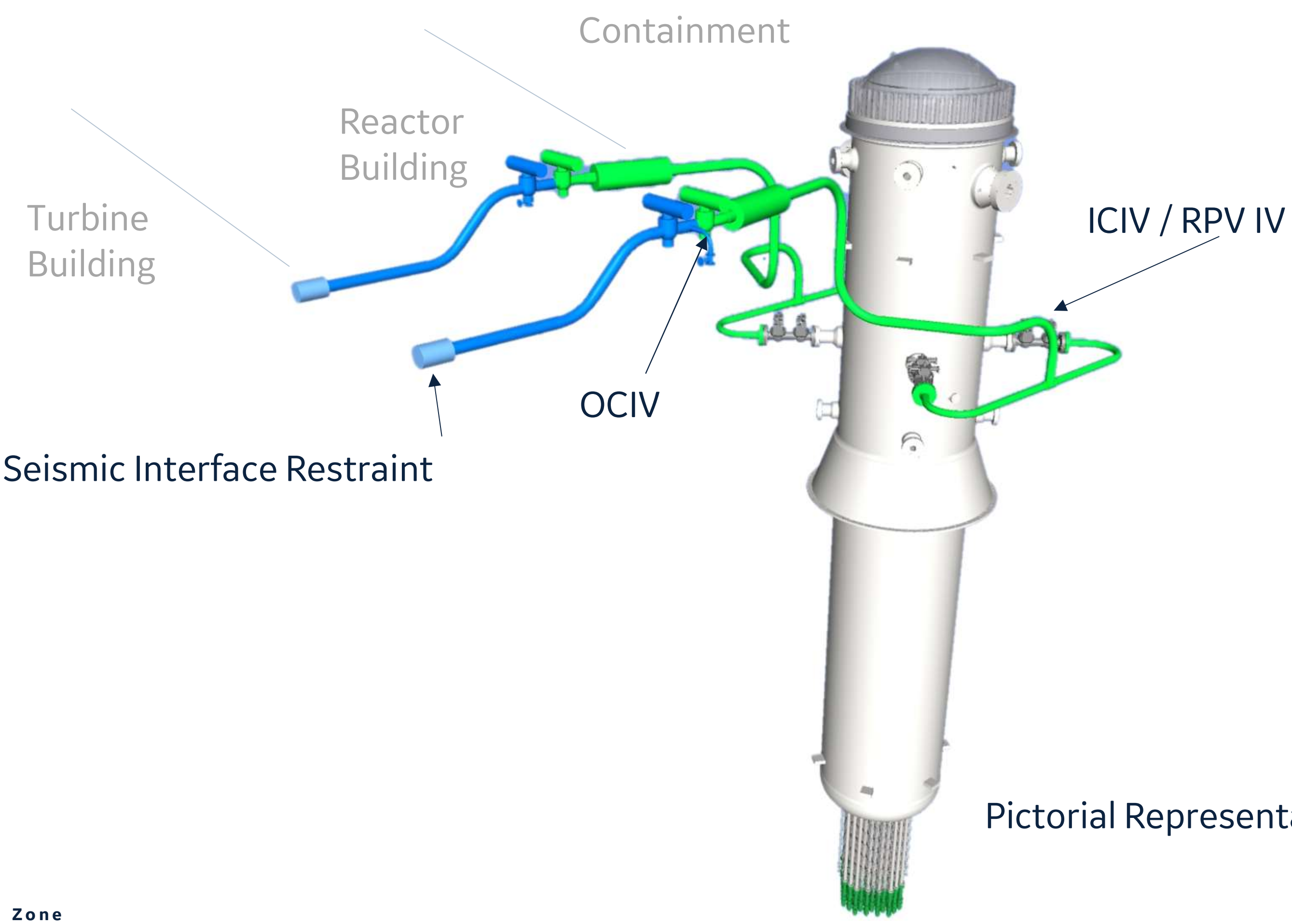
BWRX-300 Reactor Water Cleanup - BEZ



Pictorial Representation Reactor Water Cleanup

Non-Proprietary Information

BWRX-300 Feedwater Lines - BEZ



Pictorial Representation of Feedwater Line

Annex B

HITACHI

We build on our legacy, boldly innovating to provide reliable carbon-free power to the world.

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Annex C

NRC Summary of Meeting, December 14, 2022

“Safety Strategy for the BWRX-300 Reactor”

U.S. NUCLEAR REGULATORY COMMISSION SUMMARY OF THE DECEMBER 14, 2022, PUBLIC OBSERVATION MEETING TO DISCUSS PRE-APPLICATION LICENSING WHITE PAPER AND TOPICAL REPORT ON SAFETY STRATEGY FOR THE BWRX-300 SMALL MODULAR REACTOR

Meeting Summary and Staff Feedback

The meeting commenced on December 14, 2022, at 10:00 a.m. with the NRC staff's opening remarks that described the pre-application "White Paper," process as a means for the NRC staff to gain understanding of the objectives and provide early feedback on the approach the applicant will propose in a future submittal of a Licensing Topical Report (LTR) on the topic of "Safety Strategy" for the BWRX-300 small modular reactor (SMR). After the introduction of the NRC and Canadian Nuclear Safety Commission (CNSC) principal staff and review team, GE Hitachi Nuclear Energy Americas, LLC (GEH) used the "White Paper," submitted to the NRC and CNSC on December 7, 2022 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML22341A058), as a pre-submittal overview of its planned LTR, "Safety Strategy (NEDC-33934)" that is expected to be submitted for NRC staff review in early calendar year 2023.

GEH used the publicly available "White Paper," for its presentation to the NRC and CNSC staff, who participated in the meeting as part of a 2019 memo of cooperation with the NRC on advanced reactor and SMR technologies and as outlined in the September 2022, "Collaborative Information Sharing Charter," on the review of the BWRX-300 (ML22284A024). The presentation began with GEH clarifying its intent and goals associated with submitting the "White Paper" as a prelude to a possible future LTR. GEH further outlined their needed alignment aspects as they relate to their design's overall safety philosophy and design process used in development of the BWRX-300 that utilizes principles of layered defense-in-depth (DID) for the design of safety systems consistent with International Atomic Energy Agency (IAEA), SSR-2/1, "Safety of Nuclear Power Plants: Design." However, GEH specifically stated during a previous public meeting dated June 29, 2022 (ML22215A081), that after implementation of its final design and submitting it for staff's review under Title 10 of the *Code of Federal Regulations* (CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," or Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," the BWRX-300 would expect to meet all the applicable NRC regulations and guidance as well as meet the design requirements in CNSC REGDOC-2.5.2.

Throughout the meeting, NRC and CNSC staff asked clarifying questions in addition to providing constructive feedback to GEH on their Safety Strategy design framework. There were no questions or concerns from the public. The presentation from GEH did not contain any proprietary information so a closed portion of this meeting was not necessary. As a result, the meeting was adjourned at 12:45 pm.

Staff feedback and Comments on the "White Paper"

NRC and CNSC staff provided verbal feedback to GEH regarding the details and information that could be provided to enhance their proposed future LTR regarding the BWRX-300 "Safety Strategy."

- **BWRX-300 Safety Strategy Design Process and Philosophy**

GEH presented the design process and philosophy for its BWRX-300 small modular reactor (SMR) referred to as the "Safety Strategy." The objective of this process is to establish a design with a high-level of safety using a layered defense-in-depth (DID) concept using an iterative risk informed process aligned with design requirements using selected guidance of the IAEA's Specific Safety Requirements SSR-2/1, "Safety of Nuclear Power Plants Design." During past public meetings with the NRC staff, GEH has specifically stated that for licensing the BWRX-300 in the United States; GEH would meet the regulations prescribed in 10 CFR Part 50, or 10 CFR Part 52 with no exemptions expected and satisfy all applicable guidance including "NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), and other applicable regulatory guides as well as Commission Policy statements specifically for advanced passive nuclear power plant designs.

CNSC staff listened to the discussions during the meeting and asked clarifying questions. At that time, CNSC assessment of GEH's proposed Safety Strategy (as described in the White Paper) consisted of feedback to GEH that the proposed strategy appeared to be generally consistent with CNSC's regulations and processes; and that they would give a more detailed assessment at the scheduled follow-up meeting

However, the NRC staff's assessment of the BWRX-300 Safety Strategy (as described in the White Paper) consisted of feedback that GEH should provide additional information to adequately address all elements of the NRC's regulatory framework including risk informed performance-based decision making which is based on regulatory compliance, maintenance of safety margin, and treatment of uncertainties. The NRC staff also reiterated that while this novel approach by GEH could be successful, NRC approval of the safety strategy will be based on an applicant showing conformance to NRC regulations or justifying applicable exemptions.

For example, GEH in its White Paper describes a risk-informed and performance-based approach using line of defense concepts supported by probabilistic risk assessment (PRA) fault sequence frequency to identify different categories of events which may not align with the requirements for structures systems and components (SSCs) under 10 CFR Part 50 or 52. This proposed approach has the potential to result in the final BWRX-300 design which is not in compliance with NRC regulations. In addition, consistent with staff requirements to SECY-98-144, on risk-informed and performance-based regulation, the identification and quantification of uncertainties needs to be addressed in a risk informed performance-based application. The treatment of uncertainties is not discussed in the White Paper. Also, the BWRX-300 Safety Strategy would need to demonstrate consistency with the Commission's policy statements on the use of PRA in regulatory activities (60 FR 42622; August 16, 1995), severe accidents regarding future designs and existing plants (50 FR 32138; August 8, 1985), and the several staff requirements memoranda for advanced light-water designs. Potential inconsistencies or deviations will result in additional NRC resources to ensure regulations are met, exemptions are justified, and Commission's expectations are addressed.

Further, GEH identified the fundamental safety functions for Defense Line 3 with a fault sequence frequency between 1.E-2/year and 1.E-5/year. The NRC staff commented that this approach of providing numerical cutoff frequencies to delineate event categories may exclude consideration of some hypothetical design basis events required by 10 CFR Part 50. For example, if the final design is determined to have a reactor coolant pressure boundary break frequency less than 1.E-5/year, the proposed strategy could result in a loss-of-coolant accident (LOCA) being defined as a beyond-design-basis accident which would not comply with 10 CFR 50.46. A LOCA is a postulated accident that is required to be analyzed regardless of frequency of occurrence. The NRC staff also provided the postulated accidents of steam-line rupture and a rod drop accident as additional examples of non-mechanistic events that are required to be analyzed as design basis accidents. The NRC then summarized this portion of the meeting by recommending that the proposed Safety Strategy concept must comply with the NRC regulations, or if not, exemptions to specific NRC regulations should be identified.

Next during the meeting, the NRC staff noted that the proposed BWRX-300 Safety Strategy may also need to align better with the NRC regulations and NRC guidance on the characterization of the safety-related SSCs needed for the mitigation of anticipated operational occurrences (AOOs) as defined in 10 CFR 50.2 and as implemented in accordance with the guidance from SRP Chapter 15, "Transient and Accident Analysis." Specifically, in Section 3.2, "Defense Line 2," on page 18, of the White Paper, GEH states that, "there is no regulatory basis for asserting that AOOs must be mitigated by safety-related SSCs." However, 10 CFR 50.2 states that safety-related SSCs are those that are relied on during or following a design basis event to assure, in part: (1) The integrity of the reactor coolant pressure boundary, and (2) The capability to shut down the reactor and maintain it in a safe shutdown condition. AOOs are considered design basis events and are defined in Appendix A to 10 CFR Part 50, as those conditions of normal operation that are expected to occur one or more times during the life of the nuclear power unit.

The NRC staff additionally commented that the general design criteria (GDC) in 10 CFR Part 50, Appendix A provides the minimum requirements and criteria for maintaining the integrity of the reactor coolant pressure boundary, and for shutting down the reactor and maintaining it in a safe condition for AOOs and postulated accidents such that there is reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. For AOOs, the GDCs prescribe a safe shutdown condition to be one where decay heat is being sufficiently removed and the fuel integrity barrier is maintained by demonstration of appropriate margin to the specified acceptable fuel design limits. SRP, Section 15.0, states that "*the reviewer verifies that the applicant has specified only safety-related systems or components for use in mitigating AOO and postulated accident conditions and has included the effects of single active failures in those systems and components.*" This statement was specifically added in 2007 to align with the minimum requirements in the GDC discussed above.

GEH stated during the meeting that the BWRX-300 Safety Strategy is a holistic approach to classifying SSCs. Upon assessment of the proposed information, the NRC staff noted that 10 CFR Part 50, Appendix A, GDC 1, "Quality Standards and Records," require SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Further, 10 CFR 50.55a, "Codes and Standards," provides the specific requirements for design, fabrication, erection, and testing standards for certain systems and components of boiling- and pressurized-water reactors. NRC RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," describes the quality standards for SSCs acceptable to the NRC staff for satisfying the requirements of GDC 1. In the NRC staff's opinion, the White Paper neither defines the various safety classes included in the strategy, nor describes the corresponding regulatory treatment of SSCs. The proposed holistic approach to classifying SSCs for the BWRX-300 may cause inconsistency with NRC regulations and does not specifically address the Commission policy regarding advanced passive light-water reactors SSCs designated for special regulatory treatment (RTNSS).

Further, IAEA SSR-2/1, which provides the foundation for GEH's safety strategy for BWRX-300, focuses both on safety and the environment. The NRC staff commented that the White Paper did not discuss how GEH's Safety Strategy addresses the environmental aspects to conform with NRC requirements. The NRC's licensing process requires an applicant to evaluate Severe Accident Mitigation Design Alternatives (SAMDAs), which provides a systematic assessment using established guidance to examine the residual risk and if incorporation of additional mitigation is practicable to implement. It is unclear to the NRC staff if and/or how the proposed Safety Strategy considers SAMDAs to address environmental aspects identified in SSR-2/1.

- **Additional specific issues raised by the NRC staff from information presented in the Safety Strategy White Paper, include:**

  a. The White Paper describes the ultimate design goal to control the radiation exposures and restrict the likelihood of a loss of control over a nuclear reactor core, etc. However, it does not provide details on how to address the Commissions quantitative safety goals such as core damage frequency and/or large dose release frequency and their connection to the safety strategy ultimate design goal.

  b. The detailed technical basis for the numerical threshold demarcating the boundary between design basis events (Defense Line 3) and design extension conditions (Defense Line 4) as described under White Paper Section 3.3, "Defense Line 3" is not provided.

  c. The technical basis for how design basis hurricanes, hurricane missiles, and tornadoes (which are assessed at 1E-7 annual exceedance frequency) should be better aligned with the NRC regulations in addition to how GEH evaluated them using the numerical thresholds demarcating the Safety Strategy defense lines.

  d. The Safety Strategy did not seem to include provisions for or references to meeting the mitigating strategies rule under 10 CFR 50.155, "Mitigation of beyond-design-basis events." This includes the provisions related to the Spent Fuel Pool level monitoring and cooling makeup capabilities.

  e. GEH's use of numerical screening thresholds for Defense Line 5 and the concept of "practical elimination of large releases" should be reevaluated and enhanced because these thresholds could be unnecessary for the purposes of the Safety Strategy review for the NRC and, if included, could lead to a significantly expanded scope of review.

  f. A detailed roadmap explaining how the proposed safety strategy addresses NRC regulations would be valuable.

g. CNSC staff raised the comment about potential lack of independence between the defense lines due to sharing of SSCs between Defense Line 2 and Defense Line 4a, which was identified during the BWRX-300 Vendor Design Review.

- **Additional specific issues identified by the CNSC staff will be provided at the scheduled follow-up meeting.**

In summary, CNSC staff listened to the discussions during the meeting and asked clarifying questions. At that time, CNSC's assessment of GEH's proposed Safety Strategy (as described in the White Paper) consisted of feedback to GEH that the proposed Safety Strategy appears to be generally consistent with CNSC's regulations and processes. However, based on the available information, the NRC staff noted that the Safety Strategy concept, as currently proposed, could result in potential inconsistencies with Part 50 and Part 52 regulations in terms of event categorization, mitigation, and safety analyses acceptance criteria. The NRC staff additionally noted that a detailed roadmap explaining how the proposed safety strategy addresses NRC regulations could be valuable as a roadmap could help to identify any potential gaps/differences and areas that would need exemptions from the current NRC regulatory requirements. Furthermore, the NRC staff commented that it could be beneficial to see examples of implementation of various aspects of the Safety Strategy and a summary of how the proposed BWRX-300 Safety Strategy is similar to or different from (i.e., as comparison) the strategies implemented by GEH for the NRC approved Economic Simplified Boiling-Water Reactor design.