KTH Royal Institute of Technology
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Safety Issues of the ESBWR
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Nuclear Power Safety SH2612/FSH3773 VT25

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1 Introduction

In this analysis, the group members have the following roles:

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- Regulator: Caterina Armasaru
- Vendors: Eduard Corghencea & Johan Silfverplatz
- Utility: Federico Pati

The reactor design of the Economical Simplified Boiling Water Reactor (ESBWR) will be reviewed, and key safety issues for the public will be identified by the Concerned Party and the Regulator. Afterwards, the safety questions that were identified will be addressed by the Vendor and the Utility.

All the information presented in this section was taken from the ESBWR Plant General Description [1].

1.1 Reactor Design

The Economical Simplified Boiling Water Reactor (ESBWR) is a Gen III+ Boiling Water Reactor (BWR) developed by GE-Hitachi Nuclear Energy (GEH). The reactor has a thermal power of 4500 MWt and a net electrical output of approximately 1535 MWe. The reactor shares many similarities with already operating BWRs, especially its predecessors from GEH, with the addition of passive nuclear safety systems.

The Three Mile Island accident in 1979 set a spark for new innovative reactor designs that could incorporate passive nuclear safety features and in that way reduce the risks associated with manual operator actions. The response from GEH came in the form of the Simplified Boiling Water Reactor (SBWR) design in the early 1980s. GEHs goal with the SBWR was to design a simpler and more reliable design that used less components that could house passive safety functions that were independent from any external generators. This initiative was supported by the U.S. Department of Energy (DOE), Electric Power Research Institute (EPRI), and various other U.S. utilities, something that made it possible for GEH to conduct extensive evaluations of the innovative safety systems like the Gravity-Driven Core Cooling System (GDCS), Depressurization Valves (DPV), and the Passive Containment Cooling System (PCCS). Ultimately, the SBWR certification was suspended as it was deemed to be too small to be economically competitive relative other reactor designs. However, the experience gained from the passive safety systems in the SBWR would not go to waste.

Utilizing the design of the SBWR, GEH developed the design of a new larger passively safe reactor named the Economic Simplified Boiling Water Reactor (ESBWR). By using a larger vessel adopted from another GEH design, the Advanced Boiling Water Reactor, the ESBWR was able to maintain the passive safety features from the SBRW while incorporating proven BWR and ABWR design and features. The result was a simpler, safer and more cost-effective reactor that ultimately was approved by the The U.S. Nuclear Regulatory Commission (NRC) in 2014.

In Figure 1, an overview of the safety systems in the ESBWR can be observed. The design features a large cylindrical, steel-lined concrete structure enclosing both the dry-well and the wet-well. Key components to the safety such as the elevated Gravity-Driven Cooling System (GDCS) water pools and suppression pools are marked in Figure 1.

1.1.1 Vessel and internals

The ESBWR Reactor Pressure Vessel (RPV) contains a number of enhancements that improve the general safety and efficiency of the reactor. The double thermal sleeve for the feed water nozzle helps improve the durability by protecting against high-frequency thermal changes. Moreover, by including three feed water nozzles per feed water line, the incoming feed water is spread out more evenly in the vessel. Steam nozzles are equipped with flow restrictors in order to control the steam release and limiting the blowdown in choked flow conditions while

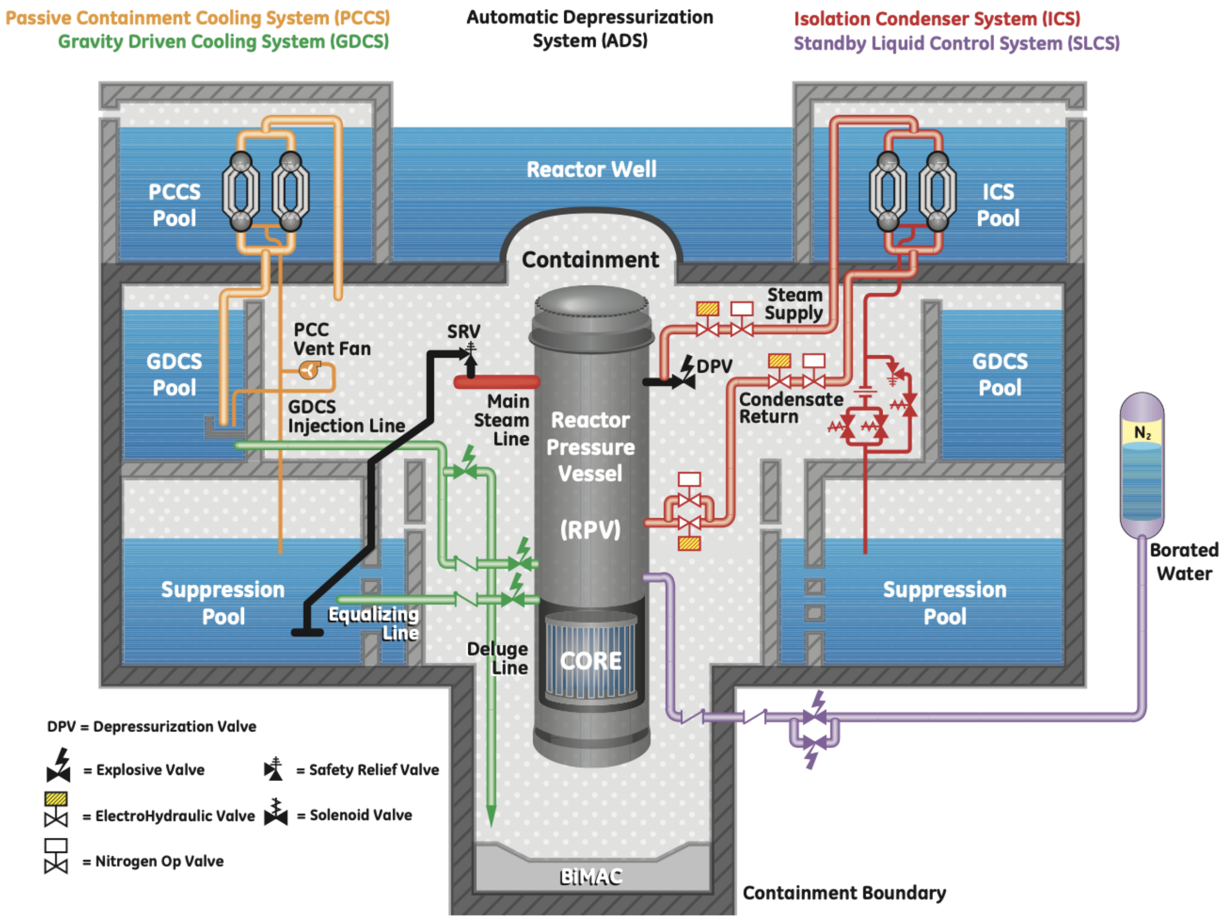

Figure 1: ESBWR Safety Systems

also making it possible to do general measurements of the steam flow. No large nozzles are present below the core in order to eliminate potential mechanically weak points, improving the overall safety.

Additionally, instead of welded plates, the RPV uses forged shell rings around and below the core, removing 30% of welds in a critical area. Of course, fewer welds improves the overall structure and reduces the risk of potential long-term cracks an leaks. This also reduces the frequency of inspections which is also great from an economic perspective. In the reactor, there is also a tall, partitioned chimney that separates groups of fuel bundles, smoothing out the water and steam flow by canceling out unwanted swirling that usually present themselves in larger open core designs.

Above the core inside the RPV, there is a large volume filled with water, acting as a built-in internal reservoir. In the event of a feed water interruption or Loss Of Coolant Accident (LOCA), the internal reservoir acts as an extra buffer against core overheating which in turn provides extra time for the operators to act on the problem and restore the normal cooling conditions. In addition, there is also a Isolation Condenser (IC) installed in the water pool above the vessel to further protect against overheating. The IC uses natural circulation to remove excess heat, preventing the need for the pressure relief valves to open, eliminating the potential release of reactor coolant into the suppression pool.

1.1.2 Reactor Pressure Vessel - Natural Circulation

The ESBWR employs passive safety features and a natural circulation primary system. By shortening the active fuel length, adding an approximately 9-meter-tall chimney above the core, and increasing the reactor vessel height, the ESBWR eliminates the recirculation system and relies entirely on natural circulation for core cooling.

The chimney, a stainless steel cylinder bolted to the top guide, supports the steam separators and provides the driving head necessary for sustaining natural circulation. The taller reactor vessel and partitioned chimney improve natural circulation without additional cost, while the taller downcomer annulus provides extra driving force and serves as a large water inventory in the event of a Loss-of-Coolant Accident (LOCA).

Steam within the chimney also dampens void collapse during pressurization transients, reducing the likelihood of Safety Relief Valve (SRV) discharges. The core consists of conventional BWR fuel bundles, though shortened from 12 ft to 10 ft to lower pressure drop and improve stability. Additionally, the absence of internal pumps or jet pumps in the downcomer minimizes flow resistance, further supporting natural circulation.

Operating experience with early natural circulation BWRs demonstrated their stability and lack of unusual instrument noise. Power adjustments were made through control rod movements, which is also the case for the ESBWR, but with electrically driven control rod drives that allow for finer positioning.

Present-day BWRs can operate at approximately 50 % of their rated power in natural circulation mode. However, stability concerns generally prevent steady operation in this range. Recirculation pump trip events in operating plants have led to a natural circulation state at around 50 % power, with plant behavior well predicted by the computational models used for ESBWR performance analysis.

Key operating parameters such as power density, steam quality, void fraction, and void coefficient remain within the established range of operating plant data. Compared to jet pump BWRs, the ESBWR features lower power and flow per fuel bundle but maintains a similar power-to-flow ratio to an uprated BWR operating under Maximum Extended Load Line Limit Analysis-Plus (MELLLA+) conditions. This ensures comparable core exit steam quality.

In summary, natural circulation is a well-established technology that offers multiple benefits. Eliminating recirculation pumps and associated hardware enhances safety, reduces maintenance, and removes transient flow instabilities. The larger reactor pressure vessel complements passive Emergency Core Cooling Systems (ECCS), improving transient response and increasing safety margins.

1.1.3 Reactivity control

Reactivity control is performed via different channels:

- movement of cross-shaped control rods containing the neutron absorber B_4C and actuated by the Control Rod Drive (CRD) system;
- presence of the burnable absorbers Gd_2O_3 in the fuel to balance the excess of radioactivity at BoL;
- as a backup, injection of borated water by the Standby Liquid Control (SLC) system.

The CRD consists of three key components:

- Fine Motion Control Rod Drive (FMCRD), which allows both for the accommodation of small power changes and the initiation of the reactor SCRAM through hydraulic insertion into the core;
- the Hydraulic Control Units (HCUs), with each of the HCUs providing the hydraulic power to carry out the scram - using high-pressure water - to two FMCRDs;
- the Control Rod Drive Hydraulic (CRDH) mechanism, which supplies high-pressure water to the HCU scram accumulators and provides purge water flow to the FMCRD to ensure reliable operation during reactivity control and emergency scram events.

1.2 Safety features of the reactor

1.2.1 Isolation Condenser System (ICS)

The Isolation Condenser System (ICS) plays different key role:

- Helps to mantain the reactor pressure stable in order to prevent the activation of Safety Relief Valves (SRVs) following an isolation of the main steam lines;
- It conserves sufficient coolant volume to avoid automatic depressurization due to a decrease in the water level during transients;
- In case the normal heat removal system is unavailabe, ICS removes both excess sensible heat and core decay heat even at high pressure, ensuring minimal coolant loss;
- ICS is also designed as a safety-related system to effectively removing decay heat after reactor shutdown while also preventing unnecessary depressurization and the activation of the Emergency Core Cooling System (ECCS).

The system consists of four independent passive loops, each equipped with an Isolation Condenser (IC) heat exchanger that facilitates steam condensation inside the pipes. Heat is transferred to a large Passive Containment Cooling System (PCCS) pool located just outside the containment structure, which is vented to the atmosphere for efficient heat dissipation. The steam line connecting the reactor vessel remains open during normal operation, while the condensate return line remains closed, ensuring that the IC, drain piping, and inline water storage tank remain filled with subcooled condensate.

When the IC is activated, the condensate return valve opens, allowing condensed water to drain back into the reactor vessel. As steam condenses inside the IC tubes, the steam-water interface moves downward, ensuring continuous heat transfer. The condensed water flows back into the reactor by gravity, efficiently removing excess heat while maintaining a stable coolant level.

To maintain system efficiency, a vent line connected to both the upper and lower IC headers helps remove non-condensable gases. A purge line is also included to prevent the accumulation of hydrogen or air in the IC steam supply line, ensuring that non-condensable gases do not interfere with system startup. Additionally, containment isolation valves are installed on both the steam supply and condensate return lines to enhance safety. A loop seal is integrated into the design to prevent leakage during standby conditions and ensure that steam flows properly into the IC without countercurrent movement back up the condensate return line.

The Isolation Condenser pool is designed with a large capacity, allowing it to remove reactor decay heat for up to 72 hours without the need for external water replenishment. This passive cooling capability enhances the reactor's safety and resilience in emergency situations. The system is automatically activated in response to several conditions, including high reactor pressure, closure of the Main Steam Isolation Valves (MSIV), a low Reactor Pressure Vessel (RPV) water level signal, or loss of DC power. Additionally, operators in the Main Control Room can manually initiate the system by opening the IC condensate return valve to ensure continued cooling if necessary.

1.2.2 Automatic Depressurization System (ADS)

The Automatic Depressurization System (ADS) plays a critical role in the safety of the Economic Simplified Boiling Water Reactor (ESBWR) by mitigating the effects of Loss of Coolant Accidents (LOCA). This system ensures the rapid depressurization of the Reactor Pressure Vessel (RPV), allowing low-pressure emergency core cooling systems to inject coolant and prevent core damage.

Function and Components The ADS is designed to prevent reactor overpressurization by discharging steam into the suppression pool. It consists of:

- 10 Safety Relief Valves (SRVs), that are pneumatically actuated valves that open to release pressure. They are divided into two groups with different pressure setpoints to provide staged pressure relief.

- 8 Depressurization Valves (DPVs), non-reclosing, straight-through valves that support the SRVs by rapidly reducing pressure during a LOCA. Each DPV has twice the depressurization capacity of an SRV, making them highly effective in emergency scenarios. DPVs are squib-actuated with a metal diaphragm seal. Two squib initiators trigger a booster, which in turn actuates a shearing plunger to open the valve.

When a LOCA occurs, the ADS is activated to depressurize the reactor quickly enough to allow emergency water injection from low-pressure systems. The primary source of emergency cooling is the Gravity Driven Cooling System (GDCS), which consists of four redundant divisions. This passive safety system ensures that coolant reaches the core without requiring active pumps, significantly enhancing the reactor's safety and resilience.

The squib-actuated depressurization valves are controlled by independent, battery-powered fire circuits. This redundancy ensures that at least one initiation path remains operational in case of a power failure. The rapid depressurization provided by the ADS is crucial for allowing the GDCS and other emergency core cooling systems to function effectively, preventing core overheating and potential fuel damage.

1.2.3 Gravity Driven Cooling System (GCDS)

The Gravity Driven Cooling System, together with ADS, act as ECCS for the ESBWR. It consists of 4 identical mechanical trains and four electrical divisions. Each division has 3 sub-systems:

1. A short-term cooling injection system, which supplies cooling water by gravity to compensate for the loss of RPV water inventory after a Loss of Coolant Accident (LOCA) and subsequent decay heat.
2. A long-term cooling equalizing system, which maintains thanks to gravity the coolant levels in the RPV over an extended period.
3. A deluge line provide a flooding mechanism for the lower drywell and is activated if the other two systems fail, providing a means to mitigate core melt scenarios. In such cases, the molten core may cause the failure of the lower vessel, allowing the fuel to reach the lower drywell floor. Thermocouples detect high temperatures in the drywell basemat, triggering squib valves to release water from the GDCS pool. These valves operate independently of GDCS injection valves. The deluge flow supports the BiMAC core catcher system, helping to cool and contain molten fuel, protecting the containment structure.

The Gravity-Driven Core Cooling System (GDCS) tank holds enough water to maintain at least 1 meter of water above the core for 72 hours without operator intervention in the event of a LOCA.

Water flow from each pool is regulated by pyrotechnic ECCS injection valves (squib valves), which remain open once activated. These gas-propelled shear valves are normally closed and open when a pyrotechnic booster charge is ignited.

Check valves in the connecting lines help mitigate unintended GDCS squib valve activation by preventing water loss when reactor pressure remains higher than the GDCS pool pressure plus its gravity head. Once the squib valves open, these check valves minimize RPV inventory loss.

1.2.4 Passive Containment Cooling System (PCCS)

The Passive Containment Cooling System (PCCS) is designed to remove heat from the containment structure and keep it within its pressure limits during Design Basis Accidents (DBAs). The system operates through six completely independent low-pressure loops, ensuring redundancy and reliability. Each loop contains a steam condenser, which transfers heat to the water in the Isolation Condenser/Passive Containment Cooling (IC/PCC) pool, a condensate drain line connected to the Gravity-Driven Cooling System Tank (GDCST), and a vent line that directs steam and noncondensable gases to the suppression pool. Working in tandem with the pressure suppression containment system, the PCCS can limit containment pressure to safe levels for at least 72 hours after a Loss of Coolant Accident (LOCA) without any need for additional water supply and can continue to operate beyond this period if the pool is replenished.

The PCCS condensers are housed in a large IC/PCC pool, positioned above and outside the ESBWR containment (Drywell - DW). Each PCCS condenser loop is designed to handle 11 MWt of heat removal capacity and consists of two identical modules for increased efficiency. These condensers are placed within subcompartments of the IC/PCC pool, and all subcompartments are interconnected at the lower ends, allowing for optimal utilization of the entire water inventory. This setup ensures an even distribution of heat dissipation and enhances the long-term cooling capability of the system.

As the pool water heats up, it can reach temperatures of up to 101°C, eventually producing non-radioactive steam, which is safely vented into the atmosphere. To prevent excessive moisture carryover and unnecessary water loss, a moisture separator is installed at the entrance of the vent discharge lines. The PCCS pool can be easily refilled using a clean water supply from the Makeup Water System, allowing for continuous cooling of the reactor indefinitely. However, if left unattended, the water will boil away after approximately 72 hours, requiring operator intervention to sustain long-term cooling during extended transients or accidents.

The PCCS heat exchangers, similar to the Isolation Condensers, are located in external cooling tanks outside the safety containment. The vent and drain lines from each condenser's lower header are routed into the Drywell (DW) through a single containment penetration per condenser module, ensuring an efficient and controlled release of steam and gases. This passive system significantly enhances the safety and reliability of the ESBWR design by reducing reliance on active safety systems while providing long-term containment cooling in case of emergencies.

1.2.5 Basemat internal Melt Arrest Coolability (BiMAC)

In the event of an accident involving the melting of the core, is not feasible to retain the molten corium inside the vessel due to the Control Rod Drive Mechanism (CRDM) penetrations through the lower head of the reactor vessel. Thus, the system relies on an alternative mitigation approach, where the molten core would be effectively quenched by flooding from the Gravity-Driven Cooling System (GDCS) tanks via deluge lines, as anticipated in the previous paragraph, ensuring the core material is rapidly cooled and controlled. With this approach risk of a more catastrophic event are minimized by managing the temperature and movement of molten material.

Moreover, the molten material leaking through the vessel will not come into contact with critical components but will instead fall onto a sacrificial concrete shield, known as the core catcher, which is positioned on the containment floor. This shield is specifically designed to absorb and cool the molten material, preventing further damage to the containment structure and reducing the potential for more severe consequences. The core catcher serves as an essential component in the accident mitigation strategy, ensuring that any molten material is safely managed and contained.

2 Discussion

In this section, the discussion between the Regulator, the Concerned Party, the Vendor, and the Utility of the ESBWR is presented.

2.1 Past LWR Experience - Regulator

The ESBWR program is stated to have a rich legacy of design, development and analysis work obtained from the earlier SBWR and ABWR projects that contribute to the development and assessments of the ESBWR [1]. Using preceding programs to develop new ones is of course desirable since it utilizes the experience gained from the old designs to make innovation and overall improvements. However, it should be noted that past evaluations cannot alone be used to justify similar new designs.

Regulators argue that while the ESBWR is based on principles inherited from predecessors like the SBWR and ABWR, it incorporates substantial passive safety changes, making direct application of the past LWR operational data unreliable.

The principle of natural circulation rather than forced coolant flow as well as the Passive Containment Cooling System represent a fundamental shift from previous LWR designs, consequently introducing new failure modes and uncertainties that should be accounted for.

2.2 Past LWR Experience - Vendor

The ESBWR design does indeed build on proven features from already licensed BWRs. By utilizing established design elements such as the reactor vessel size, control rod drives, fuel design, and material chemistry, the ESBWR uses decades of operational experience to ensure reliability while also incorporating new advancements such as natural circulation and passive safety systems [1]. The use of successful components from earlier BWR designs, for example those from the ABWR and the SBWR, provides confidence in performance and minimizes the risks associated with unproven technologies.

In our case with the ESBWR, one of the strongest arguments for the reliance on established features is that it makes it possible to do incremental innovation rather than major redesigns that possibly could introduce new failure modes. For example, maintaining the reactor pressure vessel size within ABWR parameters ensures the structural integrity of the ESBWR. Moreover, this also simplifies licensing and regulatory approvals since these parameters are known from operational BWRs. Additionally, the fine-motion control rod drives, previously validated in ABWRs, contribute to improvement in response to transients and allow for multiple shutdown mechanisms, reducing the reliance on a single failure-prone component [1].

To answer the regulator's concerns about potential over-reliance on these proven features in the ESBWR, the evaluations conducted on the SBWR and subsequently applied to the ESBWR are of course based on much more than just previous LWR assessments. We recognize that even though a lot of the key features of the ESBWR are the same as in old LWRs, they are implemented in a new iteration in the ESBWR and should therefore be tested, analyzed and approved in order to make sure that the reactor as a whole is acceptable from a safety standpoint. For this reason, extensive experimental programs have confirmed the functionality of the ESBWRs safety systems and the accuracy of analytical tools like the TRACG code, which was developed using operational data from existing BWRs. The importance of these tests are crucial, they serve as our empirical validation that the passive safety systems perform as expected under a range of simulated accident conditions[1].

Key experimental programs include:

- GIST, which validated the functionality of the Gravity-Driven Cooling System (GDCS) and confirmed TRACG's accuracy in simulating Loss-of-Coolant Accidents (LOCAs).
- GIRAFFE, which examined Passive Containment Cooling System (PCCS) thermal-hydraulics, including condensation behavior and system interactions.

- PANDA and PANTHERS, which further substantiated PCCS performance under post-LOCA conditions, demonstrating heat exchanger effectiveness and structural durability.
- Additional PANDA tests, including TEPSS, confirmed increased containment margin and performance when noncondensable gases were present.

These comprehensive testing programs were used to significantly enhance the credibility of the ESBWR's passive safety features. The passive design eliminates the need for operator action within the first 72 hours of an accident which is an improvement from active safety systems that usually require immediate intervention. This feature directly counters concerns about uncertainties of natural circulation by proving, through multiple tests, that the design behaves predictably under various scenarios.

Moreover, the ESBWR's probabilistic risk assessment demonstrates large safety improvements over previous designs. The core damage frequency for ESBWR is approximately 1.7×10^{-8} per reactor year, which is a tenfold improvement from the ABWR [1]. This reduction in risk is a result of the ESBWR's innovative passive systems, natural circulation, and simplified design, all of which are built upon historically proven features rather than experimental concepts.

To sum up, the ESBWR's reliance on established features is actually a strategic advantage rather than a weakness. It reduces licensing uncertainty, enhances safety margins, and ensures reliable performance by leveraging validated technologies. By carefully integrating these proven elements with innovative passive safety features, the ESBWR achieves significant advancements in reactor safety and efficiency while maintaining the robustness of traditional BWR designs.

2.3 Passive systems - Regulator

Following the IAEA definitions [2], a passive component does not need any external input or energy to operate and it relies only upon natural physical laws (e.g. gravity, natural convection, conduction, etc.) and/or on inherent characteristics (properties of materials, internally stored energy, etc.) and/or 'intelligent' use of the energy that is inherently available in the system (e.g. decay heat, chemical reactions etc.).

In first BWR's, forced circulation flow through the core has always been guaranteed thanks to five external recirculation loops. As a first step towards a simplified reactor, internal jet pumps were introduced inside the vessel of BWR/3 for driving the core flow. Thanks to these pumps, the recirculation flow is sufficiently enhanced so that only two external recirculation loops were needed reducing the associated piping, valves, pumps and large vessel nozzles. Lastly, the use of reactor internal pumps in the ABWR design took another step towards this process of simplification. Internal pumps are attached directly to the vessel itself, eliminating the need of jet pumps and the external recirculation systems at all and with them all the associated large pumps, valves, piping. How could you explain the reliability of passive system and how can be guaranteed an effective cooling of the core?

Moreover, the forced recirculation head from the recirculation pumps is very useful in controlling power and allows achieving higher power levels that would not otherwise be possible. The thermal power level is easily varied by simply increasing or decreasing the forced recirculation flow through the recirculation pumps. How is the power in the reactor controlled?

- Reliance on Natural Circulation: The ESBWR utilizes natural circulation to drive coolant flow, eliminating the need for active recirculation pumps. While this design simplifies the system and reduces the risk of pump failures, it also means that the reactor's cooling efficiency is heavily dependent on precise design parameters and operating conditions. Any variations, such as changes in reactor power output, coolant temperature, or system pressure, could potentially impact the effectiveness of natural circulation. Additionally, as the reactor components age, material degradation or fouling could alter flow characteristics, potentially compromising the cooling performance. Critics argue that these dependencies introduce uncertainties that may not be fully predictable or manageable over the reactor's operational lifespan.
- Limited Operator Control: Passive safety systems are designed to function without human intervention, which can be advantageous in preventing operator errors during emergencies. However, this autonomy

also means that operators have fewer options to influence the system's behavior in unforeseen or complex scenarios. In situations where passive systems may not respond adequately, the lack of active controls could limit the ability to take corrective actions. For instance, if a passive cooling system fails to maintain adequate core temperatures, operators might not have the means to manually adjust flow rates or activate additional cooling mechanisms, potentially leading to unsafe conditions.

- Design Complexity and Verification: The ESBWR's passive safety systems, while mechanically simpler due to fewer moving parts, require intricate design and thorough validation to ensure they function as intended under all possible conditions. Accurately modeling natural circulation, heat transfer, and other passive processes involves complex simulations that must account for a wide range of variables. Critics point out that these models may have inherent uncertainties, and without extensive empirical data, it's challenging to validate their accuracy fully. Moreover, passive systems must be precisely engineered to initiate and sustain the desired safety responses, leaving little margin for error in design or construction. This precision necessitates rigorous testing and quality assurance protocols to confirm that the systems will perform reliably throughout the reactor's operational life.

As the regulator, I raise the following issues about the ESBWR's exclusive reliance on passive safety systems, based on the following technical challenges:

- Lack of operational experience: passive systems depend heavily on natural phenomena (e.g., natural circulation), which lack sufficient operational and experimental data for validation. This creates epistemic uncertainties in predicting their behavior during accidents
- Model validation challenges: thermal-hydraulic (T-H) codes (e.g., RELAP, CATHARE) used to simulate passive systems have inherent uncertainties in correlations, nodalization, and failure criteria. These uncertainties propagate into reliability assessments, potentially masking critical failure modes. The ESBWR's design hinges on passive systems like the Isolation Condenser and Gravity-Driven Cooling System. If T-H codes inaccurately model their behavior and reliability assessments depend on unverified assumptions, the reactor's safety case becomes vulnerable to unanticipated failures.
- Human Factor Paradox: while passive systems reduce human intervention, operational testing and maintenance still require human actions. The interaction between passive systems and human error remains unaddressed.
- Unresolved licensing criteria: passive systems face unique licensing hurdles due to their reliance on "natural forces" with poorly characterized failure modes. Regulatory bodies may demand extensive validation data, delaying deployment ([18]).
- Hidden costs of uncertainty: passive systems eliminate active components (e.g., pumps) but introduce new costs for uncertainty reduction (e.g., advanced modeling, experimental validation). This undermines claimed economic advantages. Active systems have well-understood failure modes (e.g. pump failures),while passive systems introduce novel risks (e.g., natural circulation failure). Without rigorous comparative assessments, the safety margin of passive systems remains unproven.

2.4 Passive systems - Vendor

2.4.1 Natural Circulation

The ESBWR represents the ultimate refinement in reactor simplification by utilizing a taller vessel and a shorter core to achieve natural circulation without the need for pumps.

Natural circulation in Boiling Water Reactors (BWRs) has been a proven technology since the early days of GE BWRs, with small plants like Dodewaard (183 MWt) and Humboldt Bay (165 MWt) successfully demonstrating its feasibility. These early reactors provided valuable operational experience, proving that natural circulation BWRs could operate reliably and stably. Despite this success, GE later transitioned to forced circulation to support higher power ratings within more compact pressure vessels, influenced in part by fabrication constraints at the time. Today, GEH has returned to natural circulation with the ESBWR.

By eliminating recirculation pumps and associated components, the ESBWR design is significantly simplified. This approach enhances safety through a passive Emergency Core Cooling System (ECCS) and avoids safety/relief valve (SRV) openings during pressurization transients. The taller pressure vessel, which is necessary for these safety features, naturally enhances circulation without additional costs. The key structural change is the incorporation of a partitioned chimney above the core and a taller downcomer annulus. This design increases the driving head for natural circulation while also providing a larger water inventory for Loss-of-Coolant Accidents (LOCA). Additionally, steam in the chimney helps dampen void collapse during pressurization transients, preventing SRV discharges. The absence of jet pumps or internal pumps in the downcomer reduces flow losses, further improving circulation efficiency.

Past operational experience confirms the stability of natural circulation BWRs. These reactors have shown no unusual instrumentation noise or unexpected behavior. Modern BWRs can operate at approximately 50% of rated power using natural circulation, though stability concerns prevent steady-state operation in this region. Recirculation pump trip events in operating plants have led to natural circulation conditions around this power level, and computational models used for ESBWR performance analysis have accurately predicted these operating conditions.

For ESBWR analysis, GEH employs the TRACG code, a state-of-the-art computational tool developed and validated over decades. The code's reliability has been reinforced through extensive testing, including experiments conducted at Ontario Hydro that focused on void fraction measurements in conditions representative of ESBWR operation. These tests provided critical data for validating TRACG's ability to model flow regimes and void fractions within the partitioned chimney. Uncertainties in natural circulation flow calculations are minimal—approximately 3%—and are readily accounted for in the design process. Comparisons with pump trip events in operating BWRs, such as those at Oyster Creek, Hatch, and LaSalle, confirm the accuracy of TRACG's predictions.

Additional validation of ESBWR stability comes from the FRIGG facility in Sweden, which was designed to test hydrodynamic stability performance. The insights gained from FRIGG have been instrumental in refining the physical models and simulation tools that underpin the ESBWR's design. Importantly, the region of instability observed in other BWRs is far removed from ESBWR's operating conditions. The enhanced natural circulation flow rate in ESBWR significantly improves stability performance compared to conventional BWRs operating in natural circulation mode.

Two key factors further enhance ESBWR stability: the higher ratio of single-phase (stabilizing) to two-phase (destabilizing) pressure drop due to its shorter fuel design, and the increased ratio of the fuel time constant to the period of a stability resonance. These attributes mitigate destabilizing neutronic feedback from void perturbations, resulting in a highly stable reactor that meets conservative design and licensing criteria. Stability performance is assessed through decay ratios for channel, core-wide, and regional stability modes, with lower decay ratios indicating a more stable plant[3].

2.4.2 Reactor Power Adjustment

Since ESBWR doesn't have recirculaiton pump, it offers two ways to adjust reactor power: control rods and final feedwater temperature.

1. Control rods function similarly to those in conventional BWRs, where they are used during startup and to compensate for burnup reactivity effects. However, the ESBWR incorporates two enhancements that improve maneuverability when using control rods. First, it employs Fine Motion Control Rod Drives (FMCRD), originally introduced in the GEH ABWR. These drives use an electric motor that allows for slower movement and finer positioning compared to the conventional hydraulic system. Second, the ESBWR has the capability to move either a single control rod or a group of control rods together, providing greater flexibility in adjusting reactor power.
2. The second method of adjusting reactor power in the ESBWR involves varying the final feedwater temperature. By changing feedwater temperature, the core inlet subcooling is adjusted, which in turn affects the average void fraction in the reactor core. Increasing the feedwater temperature raises the core average void fraction, leading to a reduction in reactor power due to the negative void coefficient of reactivity. Conversely, decreasing the feedwater temperature lowers the core average void fraction, resulting in increased

reactor power.

This method of power adjustment mirrors the effect of changing the void fraction through recirculation flow in a conventional BWR. However, instead of using a power-flow map to guide operations, the ESBWR relies on a power-feedwater temperature map for power adjustments [3].

2.5 Issues with Safety Analyses - Concerned Party

According to Table 1.2-3 in the ESBWR Test and Analysis Program Description [4], a wide range of analytical assessments was performed using the TRACG code: steady-state conditions, pressurization transients, various other transient scenarios, anticipated transients without scram (ATWS), stability analysis, evaluation of the Emergency Core Cooling System (ECCS) performance during a Loss of Coolant Accident (LOCA), and the pressure-temperature response of the containment under LOCA conditions. Before the ESBWR analysis, TRACG was never used for the analysis of any phenomena other than pressurization transients, meaning that its use for LOCA and containment response analysis goes beyond its original scope. Therefore, a thorough examination of the extent and robustness of its validation is required to ensure its predictive capabilities remain reliable even under these additional conditions.

The validation methodology is described in the ESBWR Scaling Report [5] and follows a multi-tiered approach consisting of both Separate Effects Tests (SETS) and Integral Effects Tests (IETs), to establish confidence in TRACG's predictive capabilities by verifying its performance across different experimental conditions. Even though this framework seems to be comprehensive, concerns can still be raised regarding the adequacy of the scaling and the relevance of data obtained from experimental facilities.

Figure 2 presents the experimental facilities used to test the passive safety systems of the ESBWR and to validate their models in the TRACG code, together with the scale of the experimental facility with respect to the ESBWR. Indeed, while these facilities play a critical role in validation, they may not fully replicate the complex thermal-hydraulic interactions of a full-scale ESBWR system.

Table 5-1 The ESBWR Related Tests

Test	Purpose	Nominal ESBWR Scale
GIST	Integral GDCS system test	1:1000
GIRAFFE/SIT	Integral GDCS system test	1:800
GIRAFFE/He	Integral long-term containment heat removal tests with lighter-than-steam noncondensible gas	1:800
PANDA	Integral long-term containment heat removal tests	1:50
PANTHERS IC	Structural and heat transfer tests of the IC	Full-scale prototype (One module)
PANTHERS PCC	Structural and heat transfer tests of the PCC	Full-scale prototype (25% less tubes)
UCB MIT	Condensation in the presence of noncondensibles	Single-tube (near full-scale)

Figure 2: Test facilities description for TRACG validation for ESBWR. [5]

The GIRAFFE test facility was primarily used to validate the ESBWR's TRACG code as regards system interactions during the late blowdown/early GDCS phase and post-LOCA containment cooling. However, concerns can be raised regarding the facility's scaling choices. Indeed, even though the overall volume of the

RPV was preserved, the RPV regions below the core and above the MSL were shortened. This choice may not accurately replicate the fluid dynamics and thermal behaviour of the full-scale RPV, potentially affecting the fidelity of the test results. Additionally, the GIRAFFE PCC condenser used a three-tube representation of the ESBWR PCCS condensers. In tube bundles, the interaction between tubes plays a crucial role in heat exchange efficiency, and this extreme simplification to just three tubes likely leads to an overestimation of the heat removal capacity.

The testing of the Gravity-Driven Cooling System (GDCS) through the GIST facility also raises concerns. It consisted of a section-scaled simulation of the 1987 SBWR design with a horizontal area scaling of 1:1000 with respect to the ESBWR and a single pool design, which differs from the separated GDCS and pressure suppression pools of the ESBWR. Another key limitation of the GIST facility is that it is not able to replicate void formation and behaviour in the lower plenum. To compensate for the issue, the RPV water level was artificially increased, meaning that the actual void dynamics expected in ESBWR were not directly tested. This is particularly important because void formation affects GDCS actuation timing, flow rates, and re-flooding efficiency. According to [5], GEH asserts the design of the GIST facility incorporates all critical plant features capable of influencing the performance of the GDCS. However, the bigger size of the ESBWR's GDCS pools may introduce thermal stratification effects or flow resistance, which could potentially compromise the accuracy of TRACG in predicting the effectiveness of emergency cooling. Additionally, the initial conditions for testing the GDCS were computed using the TRACG code itself, which might be a questionable approach.

The PANDA facility was employed to validate the TRACG code for ESBWR by testing PCCS performance during post-LOCA containment cooling, including PCC loop operations and the delayed release of noncondensable gases. Again, some key points raise questions. For instance, the boiloff volume in PANDA is claimed to expose 50% of the condenser tube length in about 18 hours, compared to 72 hours for the ESBWR, but no proof was provided to support this claim. Additionally, PANDA tested only 20 tubes in the PCCS, whereas the ESBWR PCCS has around 660 tubes per unit - the difference in system behaviour of these two configurations might be non-negligible.

The PANTHERS tests contributed to the ESBWR qualification by providing thermal-hydraulic performance data for the PCCS and ICS condensers. However, the claim that the ESBWR PCC design can be extrapolated from the SBWR by simply extending the upper and lower headers to accommodate additional tubes is questionable. The design differences between the two systems, such as the number of tubes and system geometry, are more complex than this simplified assumption allows, which raises concerns about the accuracy of the PANTHERS data in representing ESBWR performance. Furthermore, PANTHERS collected PCC heat transfer data under steady-state conditions, justifying it on the basis that PCCS operation during a LOCA is a slow transient. This assumption may neglect critical transient effects that occur in real accident scenarios, where rapid changes in thermal-hydraulic conditions could significantly influence heat transfer behaviour, challenging the steady-state assumption.

Sections 2.5.1 and 2.5.2 present two scenarios that illustrate the consequences of potential errors that TRACG might make due to insufficient validation.

2.5.1 Delayed or Ineffective Core Cooling During a LOCA

One possible scenario that is not predicted by the TRACG code due to its limitation is a delayed or ineffective core cooling event during a LOCA. When a medium-break LOCA occurs, the RCS is rapidly depressurized by the ADS, in order to allow the LPIS such as the GDCS to refill the core. The GDCS has 8 injection valves that must open, and TRACG assumes that at least two of the eight GDCS injection valves will function [6]. However, the NRC has identified common-cause failures in GDCS squib valves and check valves as a significant risk [7]. In fact, if more than 6 valves fail due to common-cause failures, the water injection rate may not be sufficient to prevent core heat-up and the consequences could be severe. Large amounts of H_2 would be generated due to oxidation of the fuel cladding increasing containment pressure, and the core temperatures could rise beyond the validated range of TRACG, resulting in fuel degradation.

Moreover, TRACG does not fully account for hydrogen accumulation and combustion risks within the PCCS and ICS [7]. If the containment pressure is underpredicted by TRACG, hydrogen deflagration could occur, leading to localized overpressurization and containment failure if the resulting pressure spike exceeds 0.31 MPa - the containment design pressure.

Therefore, the consequences of TRACG model limitations are significant:

- hydrogen production could be faster than expected, potentially degrading the heat removal capability of heat exchangers;
- pressure buildup in the containment would be underpredicted it TRACG fails to capture the full extent of hydrogen and steam accumulation effects;
- large-scale fission product release could occur if the containment pressure exceeds its design limits.

Table 1 contains a summary of expected vs. real world outcomes for a LOCA with GDCS failure.

Table 1: Comparison of Expected vs. Real-World Outcomes for a LOCA with GDCS failure

Scenario	**Expected TRACG Outcome**	**Potential Real-World Outcome**
LOCA with GDCS Failure	Core remains covered	Insufficient cooling, rapid core heat-up
Containment Pressure Response	PCCS removes heat	Hydrogen accumulation leads to overpressure
Severe Accident Containment	Containment remains intact	Hydrogen combustion leads to breach

2.5.2 Suppression Pool Heat Removal Capacity Prediction Limitations

Among the assumptions made by TRACG there is the temperature uniformity in the suppression pool. However, thermal gradients may develop when hot steam condensates at the pool surface causing thermal stratification, which is a major concern as it reduces the effective heat removal capability of the suppression pool.

The suppression pool in the ESBWR acts as a heat sink to absorb energy from the reactor containment after a LOCA or an ADS blowdown, containing large volumes of steam and preventing containment overpressure. Together with thermal stratification, localized boiling and steam bubble formation can limit its effectiveness. If the steam flow rate is too high, boiling might occur at the pool surface, reducing the capability of condensing more steam. The steam jet submergence depth, which has not been explicitly validated in full-scale ESBWR tests, influences effectiveness of bubble condensation. [6]

Prolonged boiling in long accident scenarios leads to a reduction in the water level of the suppression pool. Long-term suppression pool depletion is not explicitly modeled by TRACG [6], which could potentially result in an underestimation of containment pressurization risks and the capability of heat dissipation is overestimated.

If the limits in modelling the suppression pool behaviour are not carefully assessed, the potential consequences are severe. Reduced pool performance leads to a containment pressure that is higher than expected and the accident progression could be accelerated. Compromisation of the suppression pool scrubbing efficiency can also occur, resulting in radioactive material bypassing containment through stuck-open vacuum breakers and uncontrolled fission product release. Additionally, the risk of combustion within containment raises due to increased hydrogen accumulation from uncondensed steam interactions [7]. Table 2 presents a summary of predicted vs. real-world possible scenarios in case of LOCA, focusing on the behaviour of the suppression pool.

Table 2: Comparison of Expected vs. Real-World Outcomes for a LOCA with Suppression Pool failure

Scenario	**TRACG Prediction**	**Potential Real-World Outcome**
LOCA with suppression pool	Stable pressure	Stratification, localized boiling
Extended suppression pool boiling	No significant water loss	Evaporation reduces cooling

2.6 Safety Analyses Adequacy - Vendor

2.6.1 TRACG Code Validation

The validation of the TRACG code for the ESBWR design has been rigorously conducted, leveraging a multi-tiered methodology that includes separate effects tests (SETS), integral effects tests (IETs), and full-scale comparisons. This comprehensive validation process ensures that the TRACG code is thoroughly examined across a broad spectrum of plant conditions, including steady-state operations, pressurization transients, ATWS scenarios, ECCS performance during LOCA, and containment pressure-temperature responses under accident conditions. Each of these tests was carefully designed to verify the model's predictive capabilities and provide robust confidence in its ability to simulate the real-world behavior of the ESBWR system. [8]

The Hierarchical Two-Tier Scaling (H2TS) methodology used in this validation provides a structured approach to assessing the adequacy of both test facilities and scaling techniques. The H2TS framework identifies the essential physical phenomena in ESBWR systems and establishes scaling criteria to ensure the accuracy of the experimental data in representing full-scale reactor conditions. This methodology has been critical in evaluating how well the scaled facilities replicate the thermal-hydraulic interactions of the ESBWR, even when perfect scaling is not possible. [9]

While the PANDA and GIRAFFE facilities operated at reduced scales, they have been designed to capture key phenomena relevant to ESBWR performance, including multi-phase flow dynamics, heat transfer, and containment cooling under LOCA conditions. In PANDA, for instance, despite operating at 1/25th of the ESBWR containment volume, vertical scaling was maintained, ensuring a valid representation of containment cooling behaviors. Although volume differences exist, the scaled facilities were designed to capture the relevant transient behaviors and multiphase flow phenomena that are critical for validating the performance of the ECCS and passive safety systems. [9]

Additionally, the use of Dodewaard reactor data, which was employed as a benchmark for validating the TRACG code, has been carefully selected for its similarity in passive safety design features, while also recognizing the differences in scale and power levels. This benchmarking allows for a qualitative assessment of the code's ability to replicate system dynamics that are relevant to the ESBWR.

Concerning the GIRAFFE facility's design, while the RPV regions below the core were shortened, the decision to preserve overall volume was made to balance test constraints with fidelity to the full-scale system. In terms of scaling the PCC condenser, while only three tubes were used to represent the full system, this design choice was deemed sufficient to capture the essential heat transfer behavior while avoiding the complexity of simulating full tube bundle interactions. This simplification allows for a focused study of key thermal-hydraulic behaviors critical to post-LOCA containment cooling. [9]

Regarding the Gravity-Driven Cooling System (GDCS) testing in the GIST facility, while the ESBWR's final design is indeed more complex than the SBWR configuration used, the GIST facility was specifically designed to test the critical features of the GDCS system. The horizontal area scaling of 1:1000 and single-pool design, while a simplification, were carefully selected to ensure a representative assessment of GDCS performance. Moreover, the GIST facility was validated against critical performance parameters, including void formation and system response under accident conditions. Adjustments to the RPV water level were made to compensate for facility limitations, but this approach still provided valuable insights into the GDCS's actuation timing and flow characteristics under simulated LOCA conditions. [9]

While concerns about scaling distortions and test simplifications are acknowledged, the validation effort has been thorough and sufficiently robust to ensure that the TRACG code is capable of reliably predicting ESBWR performance. The extensive use of blind pre-test predictions and uncertainty quantification through the Code Scaling, Applicability, and Uncertainty (CSAU) methodology further demonstrates the code's ability to accommodate real-world uncertainties and provide conservative, yet accurate, predictions. [9]

2.6.2 Consideration of the Effects of Delayed or Ineffective Core Cooling During a LOCA - Utility

The GDCS is an integral component of the ESBWR's ECCS, designed to deliver passive core cooling without active intervention. It consists of four independent divisions, each equipped with injection valves that supply cooling water to the RPV during LOCA.

While the TRACG code assumes that at least two of the eight GDCS injection valves will function during such events, it is acknowledged that common-cause failures could affect multiple valves simultaneously. However, the ESBWR design incorporates multiple divisions and redundancy within the GDCS to mitigate this risk. This design approach enhances the likelihood that at least one division will perform its intended function during a LOCA, thereby ensuring effective core cooling. Additionally, the GDCS check valves remain closed until reactor pressure slightly exceeds the GDCS injection pressure, preventing unintended water injection into the core. [10]

The TRACG code's modeling capabilities may not fully capture hydrogen accumulation and combustion risks. To address these concerns, the ESBWR design incorporates comprehensive safety features. The containment structure is engineered to withstand overpressurization scenarios, with pressure suppression pools designed to condense steam and maintain pressure within safe limits. Advanced hydrogen monitoring systems are integrated to detect and manage hydrogen levels, ensuring that any accumulation remains within safe boundaries. The reliance on passive safety systems ensures that, even without active intervention, the reactor's safety functions are maintained, reducing the risk of hydrogen-related issues. [11]

While the TRACG code is a valuable tool for simulating thermal-hydraulic behaviors during accident scenarios, it does not capture all phenomena. To enhance the accuracy of its predictions, validation of the TRACG code is done against empirical data. This is essential to ensure that its predictions align with observed behaviors under various accident conditions. Where TRACG may have limitations, supplementary analyses using other codes and methodologies are employed to better encapsulate accident scenarios.

2.6.3 Correctness of Suppression Pool Heat Removal Capacity Prediction - Vendor

Thermal stratification can occur in suppression pools when steam injection leads to the formation of a hotter water layer at the top, potentially reducing the pool's heat absorption efficiency. However, the ESBWR design incorporates several features to mitigate this effect. The reactor utilizes strategically designed steam injection systems that promote mixing within the suppression pool, thereby minimizing the formation of thermal stratification. Studies have shown that the mixing induced by these jets is a key factor in influencing the thermal response of the suppression pool, effectively preventing significant stratification.[12] Additionally, the thermal-hydraulic behavior of the suppression pool has been extensively analyzed using sophisticated CFD models and validated against experimental data. These analyses demonstrate that, under various operational scenarios, the suppression pool maintains its heat removal capability without significant impairment from thermal stratification. [13]

Concerns about localized boiling and steam bubble formation impacting the suppression pool's performance are addressed through optimized steam injection depths. The ESBWR design specifies steam jet submergence depths that have been carefully determined through both scaled experiments and analytical models to ensure effective condensation and minimize bubble-induced thermal stratification. While full-scale testing is inherently challenging, the scaling analyses provide confidence in the suppression pool's performance.[12] Furthermore, the reactor's operational protocols include continuous monitoring of suppression pool conditions, allowing for adaptive responses to maintain optimal condensation and heat transfer processes.

Addressing the potential for water level reduction during extended accident scenarios, the ESBWR's safety analysis encompasses integrated system models that account for long-term thermal-hydraulic behaviors, including water inventory management in the suppression pool. These models ensure that even during prolonged events, the suppression pool retains sufficient water levels to perform its safety functions effectively. Moreover, the reactor design includes substantial safety margins and redundant systems to compensate for any unforeseen depletion of the suppression pool, thereby safeguarding against containment pressurization risks.

Regarding the potential malfunction of vacuum breakers, the vacuum breakers in the ESBWR are designed with high reliability standards and have undergone rigorous testing to ensure proper operation under all anticipated conditions. In the unlikely event of a vacuum breaker malfunction, the ESBWR is equipped with redundant safety systems designed to manage and mitigate any resultant pressure imbalances, preventing impairment of the suppression pool's heat transfer capabilities and maintaining containment integrity.

3 Conclusion

The ESBWR represents a significant evolution in nuclear reactor design by fully embracing passive safety systems and natural circulation principles. Drawing upon the operational experience of its BWR predecessors, the ESBWR aims to improve safety margins while simplifying reactor architecture and reducing reliance on active components.

The integration of systems like the ICS, ADS, GDCS, and PCCS allows the ESBWR to withstand accidents such as LOCAs without immediate operator intervention for up to 72 hours. Furthermore, the inclusion of the BiMAC system for molten core management adds a layer of protection in severe accident scenarios.

However, despite the conceptual strengths and extensive testing that support the ESBWR's design, several concerns persist regarding the predictive capabilities of the TRACG code used in safety analysis. Notably, limitations in scaling of experimental test facilities, the simplifications inherent in thermal-hydraulic modeling, and potential gaps in validation for transient and post-accident behavior raise questions about the certainty of simulation outcomes. Scenarios involving delayed GDCS activation or suppression pool heat removal failure highlight the potential for unanticipated failure modes that could compromise containment integrity.

While the vendor has addressed many of these issues through rigorous testing and probabilistic risk assessments, the regulator and concerned parties emphasize the need for cautious validation, ongoing empirical research, and careful licensing practices. The overall safety case for the ESBWR is strengthened by its reliance on well-understood design features and passive systems; however, its novel elements must continue to be scrutinized through robust experimental and analytical frameworks.

The ESBWR embodies a promising step forward in nuclear reactor safety through passive design, yet it also underscores the importance of continuous validation and a balanced view of both innovation and its associated uncertainties.

References

[1] GE Nuclear Energy. *ESBWR Plant General Description*. Tech. rep. 2011.

[2] International Atomic Energy Agency. *Characteristics and Use of Uranium-Thorium Fuel*. Tech. rep. IAEA-TECDOC-626. IAEA, 1991. URL: https://www-pub.iaea.org/MTCD/Publications/PDF/te_626_web.pdf.

[3] GE Nuclear Energy. *NEDO-33080: TRACG Qualification for ESBWR (Non-Proprietary Version)*. Technical Report. 2002.

[4] GE Nuclear Energy. *NEDC-33079: ESBWR Test and Analysis Program Description*. Technical Report. 2002.

[5] GE Nuclear Energy. *NEDO-33082: Revision 1 "ESBWR Scaling Report (Non-Proprietary Version)*. Technical Report. 2006.

[6] GE Hitachi Nuclear Energy. *ESBWR Certification Probabilistic Risk Assessment*. Tech. rep. NEDO-33201, Revision 6, Class I. 2010.

[7] U.S. Nuclear Regulatory Commission. *ESBWR Final Safety Evaluation Report Chapter 19 - Probabilistic Risk Assessment and Severe Accidents Evaluation*. Tech. rep. ML110130034. 2011.

[8] *nrc.gov*. https://www.nrc.gov/docs/ML0225/ML022560041.pdf. [Accessed 10-02-2025].

[9] *nrc.gov*. https://www.nrc.gov/docs/ML0306/ML030650929.pdf. [Accessed 10-02-2025].

[10] U.S. Nuclear Regulatory Commission (NRC). *TRACG Application for Emergency Core Cooling Systems / Loss-of-Coolant Accident*. 2011. URL: https://www.nrc.gov/docs/ML1102/ML110280325.pdf.

[11] U.S. Nuclear Regulatory Commission (NRC). *Enclosure 3 - Control of Hydrogen During Severe Accidents*. 2016. URL: https://www.nrc.gov/docs/ML1600/ML16007A181.pdf.

[12] R. E. Gamble et al. “Pressure suppression pool mixing in passive advanced BWR plants”. In: *Nuclear Engineering and Design* 204.1-3 (2001), pp. 321–336. DOI: 10.1016/S0029-5493(00)00363-0. URL: https://ui.adsabs.harvard.edu/abs/2001NuEnD.204..321G/abstract.

[13] I. Gallego-Marcos, D. Grishchenko, and P. Kudinov. “Thermal stratification and mixing in a Nordic BWR pressure suppression pool”. In: *Annals of Nuclear Energy* 132 (2019), pp. 442–450. DOI: 10.1016/j.anucene.2019.06.002. URL: https://ui.adsabs.harvard.edu/abs/2019AnNuE.132..442G/abstract.