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Oral Presentation

Written submission from the Passamaquoddy Recognition Group Inc.

Exposé oral

Mémoire du Passamaquoddy Recognition Group Inc.

Regulatory Oversight Report for Canadian Nuclear Power Generating Sites: 2022 and Mid-term update for Ontario Power Generation's Pickering Nuclear Generating Station

Rapport de surveillance réglementaire des sites de centrales nucléaires au Canada : 2022 et Rapport de mi-parcours d'Ontario Power Generation pour la centrale nucléaire de Pickering

Commission Meeting

Réunion de la Commission

December 13 and 14, 2023

13 et 14 décembre 2023

Canada

Submission by the Passamaquoddy Recognition Group Inc.
PRGI PO Box 144 St. Stephen NB E3L 2XL

To the Canadian Nuclear Safety Commission
Regarding the 2022 Regulatory Oversight Report (ROR) for Nuclear Power Generating Sites

2023-10-30

Contents

Background

This submission is filed by the Passamaquoddy Recognition Group Inc (PRGI), in response to the *Canadian Nuclear Safety Commission's ("CNSC")* request for comments on the *2022 Regulatory Oversight Report (ROR) for Nuclear Power Generating Sites (NPGS)* which provides an overview of regulatory efforts related to CNSC-licensed nuclear power plants and waste management facilities in Canada in 2022. A public meeting with respect to this matter is scheduled for December 13, 2023.

In 2022, prior to our review of the 2021 ROR, PRGI asked if there was an expected format for our commentary regarding the ROR; we were encouraged by CNSC staff to discuss all concerns arising from our review of the ROR, in the format of our choice. Therefore, we acknowledge that while CNSC staff may deem some of the following commentary outside of the scope of the ROR, we believe the ROR provides a pertinent opportunity to highlight concerns and advance discussion with the Commission on areas of outstanding concern. We call attention however, to the potential need to expand the scope of the ROR, and request that this be discussed with intervenors and the results of these discussions be reported back to intervenors.

We appreciate the funding support through the Participant Funding Program which enabled our review.

Introduction – Passamaquoddy Recognition Group Inc.

Conservation is our sector, and thriving, protected indigenous ecosystems is our mission. We aim to explore our history, share our stories, and protect our past and future. We are honoured and committed to meet the challenges of tomorrow based in the teachings of yesterday.

Our goal is to help re-establish the means to coexist with nature, eliminating the struggles caused by 20th and 21st century human pressures. Our strategies utilize modern best practices, alongside traditional methods.

We foster innovative practices, principled creativity, and proactive means to help ensure our traditional ecosystems can re-establish themselves into healthy, sustainable, and thriving wildernesses.

In our tradition, authority is always accompanied by responsibility, and rights are accompanied by obligations. As an example, if we have the right to fish, that right is not ours alone: it also belongs to future generations of our people. For them to have a meaningful right to fish, there must be fish for them to catch. We have the responsibility to ensure that there will be healthy air, lands and waters for human and natural populations in the future.

Occupation of Qinusqinususitk - Place of the pointed land that extends into the ocean

Since time immemorial, the Peskotomuhkati have lived and thrived on the shores of the bountiful Bay of Fundy, including the lands and waters now and forever occupied/exploited by the Point Lepreau Nuclear Generating Station (PLNGS). For generations, medicines, foods, and teachings coming from these lands, sky and waters were available to our people until they were given the sole purpose of facilitating the PLNGS. Additionally, Point Lepreau has become the unacceptably proposed location for two small modular nuclear reactor (SMNR) technologies.

PLNGS exists within a mere 45 km from our sacred capital, Qonasqamkuk (St. Andrews) and 47km and 90km respectively from Peskotomuhkati communities of Sipayik (Pleasant Point) and Motahkomikuk (Indian Township).

Consent was never sought, nor granted from our people, for the development of the PLNGS on the shores of the Bay of Fundy.

Refurbishment of the station was completed in 2012 against our will.

Most recently, in opposition to our stated needs and offers to work together during a 3-year operating licence, (a period longer than NB Power's average licence length of 2.44 years) - Point Lepreau was instead granted a 10-year operating license by the Canadian Nuclear Safety Commission (CNSC). We believe, in part, the extended licence length was requested and authorized to enable an efficient co-siting of proposed SMNRs with PLNGS. Though we have been told time and time again that these projects and licences are separate, we have decades of experience with nuclear proponents and believe that the co-siting of these projects is essential to avoiding the Government of Canada's Impact Assessment Act, by virtue of the

Physical Activities Regulations. That is, new nuclear developments over 200MWth require an IA but, but this threshold jumps to 900MWth on existing nuclear sites. Thus, had the proposed SMNR existed outside of the bounds of the Point Lepreau site, an IA would have been required. Instead, we are now facing heightened risk and a concentration of radiological risk at one site, and an avoidance of the federal processes applicable to assess a project's impact to our rights, sustainability and future generations.

As we believe the projects (both existing and proposed) at the Point Lepreau Nuclear Generating Station site ought to be viewed comprehensively – especially given cumulative and compounding effects - we request a status update regarding the plans for ARC-100 waste at the Point Lepreau site. Specifically, are there plans to utilize the Solid Radioactive Waste Management Facility, which is licensed within the Operating License of PLNGS? Or, if another waste storage facility is proposed, how will it interact with existing licensed activities?

The Nuclear Conversation Backdrop

To preface our commentary regarding the ROR, as we did in 2022, let it be known that we struggle with the piecemeal approach utilized by nuclear proponents and government. Instead of participating in a holistic conversation about nuclear, including context, risk and consequence, we are asked to respond to specific indicators, projects and 'snapshots in time' and are discouraged to draw links between projects, either because of the project scope, and/or the limited mandate of the host of the conversation, or specific report. An example of this siloed approach was discussed in our review of the 2021 ROR (#4 in E-doc #6957534 Peskotomuhkati Nation_Community_Issues_Tracker) which states,

> "The ROR focuses on hard sciences, pure and applied - physics, chemistry, engineering - without much attention to the biomedical or ecological sciences… We found that the 2021 ROR and most other CNSC documents, unfortunately lack context for those interested in understanding whether or not the health and safety of persons and the environment is indeed being protected from nuclear-related risk. Information related to the reasons for the various CNSC regulations - the many harmful biological effects of chronic or acute exposures to radioactive materials, and the multitudinous pathways of radionuclides through the environment and through the body - the actual health threats and real environmental risks - go unmentioned…"

Since this comment was made, the CNSC staff has responded with the purpose of the ROR, so we understand that each CNSC document has a very specific goal, but nonetheless suggest this piecemeal approach is a barrier to fulsome comprehension and discussion of the nuclear 'ecosystem'.

ROR Response

We provide the following commentary which first discusses some overall recommendations, then moves onto concerns that remain from our review of the 2021 ROR and concludes with new observations.

Critical Legal Developments Should be included in the ROR

While the ROR is focused on activities and operations that occurred in 2022, in a number of instances, CNSC Staff provide updates on activities and compliance activities that occurred in 2023[1] which we appreciate. We submit that in a similar vein, developments in nuclear law and policy from 2023 (and going forward) that are directly relevant to the operation of nuclear power plants and licensees should also be required

[1] *See for instance*: 2023 update relating to Gentilly-2 on p 42; 2023 updates relating to Darlington on p 54, 61

inclusions and referenced within the text. This would also directly further the objects the CNSC to 'disseminate objective scientific, technical and regulatory information to the public' as set out in section 9(b) of the *Nuclear Safety and Control Act.*

As the licensees reviewed in the ROR have at least 10-year licences and the timing of the license renewals among nuclear power plants do not necessarily align, the ROR provides a pertinent opportunity to incorporate discussion and updates on new nuclear law and policy that ought to be reviewed by the CNSC and licensees, to verify conformance. A few examples are detailed below.

Policy for Radioactive Waste and Decommissioning

In March 2023, the Ministry of Natural Resources released the modernized "Policy for Radioactive Waste and Decommissioning" for Canada. A number of provisions are directly relevant to nuclear power plant licensees as "waste generators and owners" who "must manage radioactive waste, including its disposal, in a manner that protects human health, safety, security, and the environment over the long term."[2]

Sections which ought to have been canvassed in the ROR include sections 1.5 – 1.8, which speak to the need to categorize, minimize and ensure adequate funding for the long-term management of radioactive waste (one of our concerns discussed in our 2022 PLNGS license renewal intervention, as well as our review of both the 2021 and 2022 ROR). As the Policy states:

Waste generators and waste owners will:

[2] Canada's Policy for Radioactive Waste Management and Decommissioning, online, Our Vision [**Policy**]

> 1.5 ensure protection of human health, safety, security and the environment, and ensure nuclear non-proliferation, for present and future generations in their radioactive waste management and decommissioning activities, including transportation and disposal, and in the development and operation of their radioactive waste management facilities, locations, and sites;
>
> 1.6 ensure adequate funding is available for long term management of radioactive waste, including disposal sites, as well as the decommissioning, clean-up, remediation, and closure of these facilities and sites, as applicable;
>
> 1.7 prevent and minimize the generation, volumes and activity levels of their radioactive wastes, to optimize waste management, through appropriate facility design measures and through operating and decommissioning practices, including the recycling and reuse of materials, while taking into account health, safety, security, nuclear non-proliferation, environmental and socio-economic considerations;
>
> 1.8 follow relevant national standards to characterize, classify and record their radioactive waste inventory in order to define and implement radioactive waste management and decommissioning solutions that are commensurate with their risks in both the short and long term;

We also recommend the inclusion of sections 2.4 – 2.5, which <u>require</u> the open, accessible and transparent planning for radioactive waste. As the Policy states:

> Waste generators and waste owners will:
>
> 2.4 plan radioactive waste management and decommissioning projects in an open and transparent manner, with early and ongoing input from Indigenous peoples, provinces, territories, interested communities, including current and prospective host communities, scientific experts, and other interested persons in Canada;
>
> 2.5 ensure their communications, information, and documentation on radioactive waste management and decommissioning are easily accessible by the public, accurate, and are kept up to date in order to facilitate, among other actions, open, transparent and inclusive engagement;

We also recommend the inclusion of sections 3.7-3.10 which are specific working in partnership with Indigenous peoples and building capacity to engage meaningfully. As the Policy states:

> Waste generators and owners will:
>
> 3.7 acknowledge the unique status of Indigenous peoples as rights holders in Canada; commit to respecting their rights; and work in partnership with Indigenous peoples to gain a greater understanding of the implications of radioactive waste management and decommissioning projects on these rights;
>
> 3.8 work in partnership with Indigenous peoples to gain a greater understanding of their Indigenous knowledge and advice with regards to radioactive waste management and decommissioning projects;
>
> 3.9 demonstrate meaningful and respectful engagement, on an early and ongoing basis, with Indigenous peoples who may be affected in the siting, construction, operation, and monitoring of radioactive waste management and decommissioning projects;
>
> 3.10 commit to building capacity among Indigenous peoples to permit their meaningful participation in engagement in the planning, development, and operation of radioactive waste management and decommissioning projects.

PRGI recommends that CNSC Staff provide an update at the ROR meeting reviewing the conformance of NB Power's operations with the modernized Policy. We also request that the CNSC task NB Power with reporting on the provisions of the Policy and require an assessment of the action plan in next year's ROR. As many of the provisions directly require the open sharing of information and engagement with Indigenous peoples, PRGI expects to receive full disclosure of waste management and decommissioning plans and renewed efforts to seek our input and cooperation in decision-making.

UN Declaration on the Rights of Indigenous Peoples

The CNSC is as an agent of the Crown[3] and thus obligated with fulfilling the Honour of the Crown for consultation and accommodation. In 2021, the *United Nations Declaration on the Rights of Indigenous Peoples Act* (*UN Declaration Act*) received Royal Assent and came into force, providing a process for Canada to work together with First Nations to "implement the UN Declaration based on lasting reconciliation, healing and cooperative relations."[4]

In 2023, the "United Nations Declaration on the Rights of Indigenous Peoples Act Action Plan" (the Action Plan) was released. It contains 181 measures to contribute to the achieving the *UN Declaration Act* and sets out a number of principles to guide regulatory activities, such as:

- "ensuring First Nations have sufficient, sustainable data capacity to control, manage, protect, and use their data in order to participate in federal decision-making processes
- encouraging consultation which could lead to the setting of measures enabling the exercise of regulatory authority by First Nations" [5]

While recognizing the CNSC is not directly named in the Action Plan's proposals, its principles provide a helpful starting point to enable cooperation with Indigenous peoples in implementing the UN Declaration.

[3] *Nuclear Safety and Control Act,* s 8(2)
[4] Department of Justice, "Implementing the United Nations Declaration on the Rights of Indigenous Peoples Act," online: https://www.justice.gc.ca/eng/declaration/index.html

[5] *Ibid* at s 30 and 34

PRGI recommends that the CNSC review the sufficiency of licensee activity in light of the principles and priorities set out in the Action Plan given their relevancy to federal regulators. PRGI also recommends all RORs going forward, include assessments of licensee activity against the benchmarks set out in the Action Plan, including:

- Advancements in self-determination, including recognitions of decision-making authority held by PRGI over its lands
- Concrete actions to advance nation-to-nation relationships
- Progress on the disclosure and sharing of information to facilitate PRGI's more informed participation in decision-making

Fisheries Act Authorization for Point Lepreau Ought to be Detailed

One particularly significant gap in the ROR is the lack of discussion regarding the Fisheries Act approval that NB Power has only now been granted, despite having operated since the 1980s. Section 35 of the Fisheries Act prohibits anyone from carrying on any work, undertaking, or activity that results in the harmful alteration, disruption, or destruction of fish habitat, unless it has been approved via permit or Ministerial authorization.

As the ROR notes: "NB Power was granted a Fisheries Act Authorization by Fisheries and Oceans Canada on August 23, 2022."[6] However, this statement is not accompanied by any discussion that details the nature of the approval, its conditions or role the CNSC plays in ensuring compliance.

[6] ROR, p 114

We do not support the scarce attention provided to this authorization, especially as NB Power has operated without one, in non-conformance with the *Fisheries Act* since operations commenced in the 1980s, and it was granted *after* the CNSC made its licensing decision. Thus, this *Fisheries Act* authorization has never been openly reviewed and discussed as part of a licensing hearing.

This is particularly significant given the authorization permits:

- The death of approximately 29,900 kg (29.9 MT) of fish and macroinvertebrates per year due to impingement and entrainment

We note conditions on the authorization include:

- The condenser cooling water system withdraw no more than 33 m^3/s of water and velocity remains below 0.26m/s
- There be no change in the location of the water intake system

The authorization also requires NB Power to notify DFO within 48 hours if the conditions are not met and a corrective action plan submitted within 5 days. PRGI takes this opportunity to also request notification of any non-compliance with the authorization.

PRGI also recognizes that the memorandum of understanding (MOU) between the CNSC and DFO requires consultation and coordination with Indigenous communities in *Fisheries Act* authorizations.[7] In light of the proposed new nuclear projects at the Point

[7] Memorandum of Understanding (MOU) Between Fisheries and Oceans Canada and Canadian Nuclear Safety, Commission For Cooperation and Administration of the Fisheries Act Related to Regulating Nuclear Materials and Energy Developments, December 16, 2013, online, s 1(f), 3(a)

Lepreau site, PRGI requests to be both engaged and meaningfully involved in subsequent *Fisheries Act* authorizations.

While we recognize the CNSC may deem our comments regarding new nuclear at Point Lepreau 'out of scope' for this ROR, given the inordinate delay and decades of nuclear operations that occurred without a *Fisheries Act* permit, PRGI again insists on the need for our early and full engagement regarding new approvals *and* that it be required before any new licences at the site are granted.

Ongoing Discussion of Concerns

Last year, the Passamaquoddy Recognition Group Inc. (PRGI) reviewed the 2021 ROR. This review has been given a document number (CMD-M34.1) by CNSC staff and is available at https://nuclearsafety.gc.ca/eng/the-commission/meetings/cmd/index.cfm#meeting-20221101-20221103. Moreover, CNSC has produced a spreadsheet (E-doc #6957534 Peskotomuhkati Nation_Community_Issues_Tracker) which includes the 41 different issues, concerns and recommendations raised in the PRGI's review of the 2021 ROR, together with CNSC's response to each of these concerns (although this document is not available publicly, we invite anyone interested in reviewing it to contact PRGI).

While we appreciate the efforts of the CNSC staff to provide responses to us, many recommendations were made to improve the document for the benefit of all readers. As we have noted to CNSC staff in discussion since our review of the 2021 ROR was provided, only one of our 41 recommendations from our review of the 2021 ROR, was simply accepted – however in the 2022 ROR the section was very different from 2021 and the recommendation did not show up. Approximately 5% of our recommendations addressed information we thought to be missing from the 2021 ROR, but that was indeed included (and the CNSC staff response pointed us to the information location), approximately 10% of our recommendations were met with a note about future discussion, almost 20% were met with a note about the scope of the ROR, 29% of our recommendation were not accepted, and additional 29% of our recommendations were not accepted but met with some sort of variation on the response, “This recommendation will be taken into consideration for future RORs.”

Based on the CNSC responses, PRGI both better understands the purpose of the ROR, and concurrently finds some of these CNSC responses troubling. Since CNSC has encouraged an “on-going two-way dialogue” on matters of contention, PRGI feels it is necessary to explain why some of these responses are felt to be concerning, as many of these same issues remain unresolved in the 2022 ROR. For the sake of convenience, in this next section, we will follow the same numbering that CNSC staff has used in tracking our concerns and CNSC responses (E-doc #6957534 Peskotomuhkati Nation_Community_Issues_Tracker).

Concern #1 – Detailed Description of 2022 CNSC Decision re: renewal of PLNGS Operating License.

The detailed description of decision regarding the most recent PLNGS relicensing was released at a point after PRGI submitted their review of the 2021 ROR. CNSC staff and representatives from PRGI have since discussed some of the legal context of this decision but we do not accept that the decision aligns with recent legal developments and Canada’s international obligations respecting Indigenous rights and the need to obtain our free, prior and informed, consent.

Due to new nuclear proposals in our homeland, the CNSC ought to be able to justify its decisions in light of more recent developments, such as the UN Declaration and Canada's United Nations Declaration on the Rights of Indigenous Peoples Act (UNDA) Action Plan. At various points we have asked for CNSC legal representatives to provide interpretation of the most recent PLNGS relicensing decision, but thus far, we have instead participated in discussion with non-legal CNSC staff. Although we agree this is a good start, because of the complexity of the topic and the information overload we currently experience, we request a written legal justification by CNSC lawyers.

Concern #2 Acknowledgement of Fact

This concern remains unresolved. PRGI asked that a simple statement of fact be made in the Executive Summary of the 2022 ROR, acknowledging that the power reactors (including Point Lepreau) were built without Indigenous consent, and that those plants (including Point Lepreau) continue to produce and store long-lived toxic waste materials without Indigenous consent.

In its response, CNSC says it "appreciates the feedback" and will "consider it" for the 2022 ROR. However, such an acknowledgement is not found in the Executive Summary of the 2022 ROR. Based on this omission, we now ask for further detail; how the matter was considered, by whom and why it was decided not to make the acknowledgement?

The CNSC response found in E-doc #6957534 (Tracker) states that staff is "committed to better understanding and addressing concerns" and to providing "opportunities for meaningful long-term engagement" based on "ongoing two-way dialogue" to "better understand each other's perspectives" and "look for collaborative solutions".

We are unsure what more is necessary to "understand" about our request but are willing to discuss with the appropriate CNSC team, as we seek to understand how CNSC will "address" this concern. The lack of Indigenous consent is not subjective, it is fact. We seek to understand where the "collaborative solution" is applied in CNSC's present unilateral decision not to provide a straightforward acknowledgement of this fact? To continue the "on-going two-way dialogue" regarding this issue, we highlight that continuing to disregard the truth, means that we can never progress to reconciliation.

We continue to call upon the CNSC to acknowledge that the current state of nuclear operations at Point Lepreau were approved, developed and licensed without our free, prior and informed consent. Recognizing this history and the impact of excluding us from decisions made at the site must inform PRGI-Crown (as exercised by the CNSC) relations.

Concern #3 Continuing Operations & Reconciliation

PRGI in 2022, and now in 2023 asks how "supporting, and allowing PLNGS to continue to operate without consent on our homeland, promotes and facilitates reconciliation?" This question is still valid with respect to the 2022 ROR, as well as to the upcoming license application for the ARC-100. In E-doc #6957534 (Tracker) the CNSC responds not with any content that leads us to further understanding of the answer to our question, but with a commitment for further discussion. We are also (thus far) committed to further discussion; however, we note that either:

a> the CNSC must start to provide content, not just commitments to eventually provide content, or

b> all partners decide is it acceptable to defer content to ‘future discussion’ at a mutually convenient time.

However, If the latter is the case, we then recommend that in response to ARC-100 license application, the CNSC and proponent rid themselves of any expectation of a timeline or any specific deadlines, and instead commit only to ‘further future discussion’. We use this example, not to be facetious, but to try to elucidate for the CNSC our perspective, to advance CNSC’s understanding of our perspective.

Additionally, we feel that the non-specific CNSC responses trivialize our opposition to the continued production of long-lived radiotoxic materials within our homeland. CNSC gives no clear indication that it will take any action to ensure that no more long-lived radiotoxic materials will be produced in our homeland, which, as discussed above, is in contravention on Canada’s commitment to the UN Declaration on the Rights of Indigenous Peoples which requires our free, prior and informed consent prior to the disposal of any hazardous materials on our lands (Article 29.2).

We call upon the CNSC to ensure the Commissioners and its staff receive appropriate cultural competency training, which includes the understanding of Treaty obligations, in keeping with the Truth and Reconciliation Commission of Canada's Calls to Action. There is a positive obligation on the CNSC to seek such training and not rely on the PRGI to educate CNSC staff. For the CNSC to better address our concerns and interests, the CNSC must be properly educated and informed on Treaty, as well as the adoption of new Canadian commitments, if we are to trust them and expect them to respect and honour our relationship.

Concern #4 Inclusion of Potential Pathways and Biological Effects of Radionuclides

In 2022 PRGI asked that more information be made available on the potential pathways and biological effects of radionuclides that are or may be released during routine operations or under accident conditions at the plant. In response, CNSC staff respond with the purpose of the ROR, which is now better understood, as well as various resources we have since reviewed.

However, none of these CNSC resources discuss the pathways of radionuclides through the environment or through the human body, nor do they discuss adverse health effects that may result from human exposure to atomic radiation or to radioactive materials. In particular, there is no discussion of the hazards associated with chronic tritium ingestion or inhalation over a long period of time.

Since CNSC is responsible for protecting the health and safety of humans and the environment, but it has been determined that the ROR is not the appropriate forum for this information, and concurrently the CNSC desires a meaningful relationship with Indigenous nations, we recommend that CNSC creates and supports a forum where adequate background documentation that details why humans and the environment need protecting from these radioactive materials can be co-developed with Indigenous Nations. We recommend that the development of this communications tool would be by and for those impacted by nuclear developments and legacies. This health and environmental information should be freely available, suitably detailed, and clearly referenced in the ROR report.

Concern #5 Tritium-contaminated Heavy Water

In its 2022 submission, PRGI recommended that the Point Lepreau tritium-contaminated heavy water be replaced immediately with “clean” (i.e. non-radioactive) material. CNSC responded that NB Power has committed to replacing the tritiated moderator water with “heavy water that has a lower tritium content” in 2028. That schedule is six years after the PRGI recommendation was made in CMD-M34.1, and 16 years after the issue was first flagged (see below). The time-lag of this response calls into question the CNSC’s oft-repeated commitment to the ALARA (As Low as Reasonably Achievable) principle. As well, we are concerned with the lack of quantification regarding the term “lower tritium content”.

It is well-known that radioactive tritium, created in the moderator by stray neutrons, is responsible for most of the internal radioactive contamination of workers in the plant. It is also the major contributor to radioactive releases into the environment, both airborne and aqueous. The primary responsibility of the CNSC is to protect the health and safety of humans and the environment. Such a 16-year delay in replacing the contaminated heavy water at the Point Lepreau plant, thereby reducing radioactive exposures both inside and outside the plant, reflects poorly on the CNSC’s safety culture.

CNSC has explicitly adopted the LNT (linear no-threshold) model of radiation carcinogenesis in its regulatory role. This implies that all radiation exposures may contribute to the chance of a radiation-induced cancer later on. Therefore, even if tritium exposures and tritium releases are within regulatory limits, it does not exonerate the CNSC from taking action to keep all exposures and releases “As Low as Reasonably Achievable” (ALARA). Using non-contaminated heavy water is a reasonably achievable way of reducing both exposures and emissions, yet such action has been delayed by CNSC and by NB Power for a decade and a half, without any rationale. Is this a purely economic choice?

For the record, on February 7, 2012, letters were sent to the CEO of NB Power, New Brunswick Premier David Alward, and the President of the Canadian Nuclear Safety Commission, asking each of them to take immediate action to ensure that the radioactive (tritium-contaminated) heavy water from the Point Lepreau reactor be replaced, before any consideration was given to refuelling and restarting the reactor. These letters were sent nine months before the CNSC authorized the restart of the reactor.[8]

Nobody had known that NB Power was re-filling the calandria vessel with radioactively contaminated heavy water in the newly refurbished plant until an accidental spill of a few litres happened in December 2011. This spill created an onsite radiological emergency due to airborne tritium despite the fact that the plant itself had been shut down for almost four years (since March 2008) for refurbishment.

It even came as a surprise to the Canadian Nuclear Safety Commission (CNSC) that NB Power was refilling the reactor core with old, contaminated heavy water instead of using non-radioactive material. The CNSC President, Dr. Binder, called the situation "unsettling". Although Canada's nuclear regulator, CNSC apparently did nothing to deal with the root cause of the problem - by refusing to allow NB Power to re-use the contaminated heavy water.

The situation worsened when workers had to be evacuated due to the tritium contamination in the heavy water (much of which had become airborne) but two men were trapped inside for two hours because safety doors malfunctioned. Moreover, that radioactive

[8] See http://www.ccnr.org/Media_Release_2012_02.pdf for more information

spill was a repeat of a similar accident that had happened 15 years previously. What did CNSC do to prevent such a recurrence, and why did CNSC oversight and action not prevent the 2011 and 2022 incidents?

CNSC has a duty to ensure that all radiation exposures are kept as low as reasonably achievable. This can be done by replacing the tritium-contaminated moderator water with clean material.

PRGI requests further detail regarding the expected tritium content, and a justification regarding the schedule, and we continue to recommend that this be done immediately and not six years in the future.

Concern #6 Emergency Planning

As in 2022, we again call on the CNSC to extend the emergency planning zones (EPZ) and the Extended Planning Distance (EPD) to reflect international best practice. The establishment of these zones are a principal tool of offsite emergency preparedness and critical to safeguarding communities and our environment in the event of a radiological release.

In 2022 PRGI called attention to the fact that a severe nuclear accident at Point Lepreau could affect communities in Nova Scotia and Maine as well as those in New Brunswick. Emergency preparedness should therefore not be limited to New Brunswick but should include portions of those two additional jurisdictions. The New Brunswick Off-Site Emergency Measures Plan, however, mistakenly concludes that the EPD zone need only have a radius of 50 kilometres, citing an International Atomic Energy Agency (IAEA) document as its authority for this conclusion, but the radius is based on a misinterpretation of the IAEA document. The IAEA suggests a 50 km radius EPD zone for any reactor that generates less than 1000 megawatts of heat. For larger, more powerful reactors, the IAEA suggests a 100 km EDP zone. The NB Off-Site Emergency Measures Plan mistakenly presumed that the Point Lepreau reactor (700 MWe) fits into the first category (50 km radius) as described by IAEA, which it does not, as PLNGS generates over 2000 megawatts of heat at full power. For such a large plant IAEA suggests a 100 km Extended Planning Distance.

To generate 700 megawatts of electricity, a nuclear power plant must generate almost three times as much heat (about 2100 megawatts of heat). For this reason, one must carefully distinguish between thermal megawatts of heat (MWth) and electrical megawatts (MWe). Point Lepreau generates approximately 2180 megawatts of heat to produce about 700 megawatts of electricity.

For reactors of 1000MWth, the IAEA also recommends the Ingestion Planning Zone ought to extend to 300km. The Point Lepreau reactor has a capacity more than double this size, at 2180 MWth and thus falls within the IAEA's recommendation of a 300km Ingestion Planning Zone and yet, the current Ingestion Zone at Point Lepreau only extends 57km.

PRGI asks that these mistakes be corrected. In 2022, CNSC responded that the NB planning document was reviewed by staff as to “accident selection, dispersion modelling, dose estimates and methodology”. CNSC concluded that the use of the 50 km radius as an EPD zone is acceptable and is “in line” with IAEA guidance. However, those details – accident selection, dispersion modelling, dose estimates and methodology – are not included in the NB Off-Site Emergency Measures document, so it is unclear exactly what “review” the CNSC staff conducted, and the review is neither included nor referenced. Please provide the resources relied upon for the CNSC review.

CNSC has not acknowledged that an error was made by NB authorities in citing an IAEA document that, in itself, did not recommend a 50 km EPD zone for such a large reactor as Point Lepreau. It does not reflect well on the safety culture if misinterpretations of such a fundamental nature go unacknowledged and uncorrected. Even if the subsequent CNSC analysis (which has not been explained or delineated) indicates that the error is not consequential in this instance, it is still an error and should be acknowledged as such.

Since the Peskotomuhkati Nation extends into Maine, and since a 100 km EPD zone would require emergency preparedness to extend into Maine, PRGI requests that CNSC provide a detailed explanation of why the more limited EPD zone is considered appropriate despite IAEA guidance to the contrary.

Further, and in light of proposed new nuclear activities at the site and the heightened concentration of radiological risk this will bring, PRGI requests the emergency planning zones - at a minimum, be on par with suggested best practice by the IAEA.

Concern #7 Source Term

In 2022 PRGI questioned the “source term” (the amount of radioactivity able to be released) in the event of a severe nuclear accident at Point Lepreau. The source term has a direct relationship to the size of the Extended Planning Distance (EPD) zone discussed above under Concern #6.

CNSC responded that the source term for a severe accident at Point Lepreau dates back to a 2009 PSA (probabilistic safety analysis) carried out by NB Power long before the 2011

Fukushima triple meltdown. Four scenarios studied by NB Power led to a range of estimated radioactive cesium-137 releases from a low of 0.13 terabecquerels to a high of 1,740 terabecquerels, a terabecquerel being a million times a million becquerels. (A becquerel is one radioactive disintegration per second. Cesium-137 is one of the most intensely radioactive poisons created inside every nuclear reactor.)

These NB Power estimates of radioactivity released during an accident can be compared with the much larger releases of 80,000 terabecquerels of cesium-137 from the Chernobyl accident, and 17,000 terabecquerels of cesium-137 from the Fukushima accident. CNSC goes on to say that during an emergency, “the actual source term will be calculated based on the actual station conditions and not from pre-selected source term.” Evidently there is a very wide range of uncertainty that cannot be predicted accurately by NB Power or CNSC or anyone else.

Given the magnitude of the uncertainties in the source term, and taking into account the unpredictability of the weather conditions (i.e. which way the wind is blowing, whether it is raining or not, the possibility of a temperature inversion, etc.) PRGI reinforces their recommendation that the more conservative 100 km limit for the emergency planning distance zone be implemented. This will be in close affinity to the actual IAEA recommendation. To continue to do less is to deprive neighbouring jurisdictions of the information needed to prepare effectively.

Concern #8 Degasser Condenser Valves

During the Point Lepreau licensing hearings, and again in its comments on the 2021 ROR, PRGI questioned the adequacy of a critical safety feature, the overpressure relief valves (that are technically called degasser condenser valves). If the pressure in the primary cooling system cannot be relieved fast enough during a severe over pressurization event, some components in the cooling loop will likely burst inside the reactor. This could seriously jeopardize the cooling of the reactor fuel and exacerbate any fuel damage that may occur as well as greatly increase and facilitate the dispersal of radioactive releases from the damaged fuel.

During the licensing hearings we heard from a CNSC subject matter expert on this question who stated on the record that his team (the "Working Group") was preparing a report on this very topic - relieving any over pressurization that may occur in the primary coolant loop.

In response to our concern, in E-doc #6957534 (the Tracker), the CNSC staff responded by indicating:

"4a) The commissioners are aware with the previous reviews performed to investigate the over-pressure relief valves capacity on the degasser condenser (CMD 17-M.14/17-M.14.A – see e-Docs# 5191580 and 5150969, respectively). The latest information by the author of the recent papers was presented to a panel of pressure boundaries and safety experts within CNSC and no consensuses [sic] was reached between the papers' author and the panel. Therefore, CNSC has decided to hire an external independent reviewer who is mutually acceptable to the panel and the author to assess the subject. Based on the current knowledge, there are no safety concerns that are confirmed.

4b) As mentioned in the above response, there is ongoing review and assessment by an independent third party that is acceptable to both the papers' authors and CNSC experts."

However, PRGI has subsequently learned that there was indeed an internal CNSC briefing note on this subject produced on June 17, 2022, which amplifies our concern on this topic, and is entitled “Briefing Note on Degasser Condenser Relief Capacity for CANDU Reactors”. This briefing note – which is attached to this submission - concludes, in part, as follows:

“After significant debate and analysis, the Working Group concludes that the relief capacity of the degasser condenser is inadequate for beyond design basis accidents such as a prolonged loss of all AC powers leading to a sustained loss of all heat sinks. It does not meet the current ASME and CSA's stated design requirements. A comparison with a similar Light Water Reactor system also points to the inadequacy for beyond design basis accidents.”

This conclusion certainly seems like a consensus position from at least the Working Group, however, we remain unsure of the interaction (and membership) of the CNSC ‘Safety Expert panel’, versus the ‘external independent reviewer ‘and the ‘Working Group’. What we do understand that the loss of all AC (alternating current) power referred to in this paragraph from the briefing note is commonly called a “station blackout”. It is just such a station blackout that caused the triple meltdown at the Fukushima Daiichi nuclear generating station in Japan in 2011.

It is our understanding that nuclear fuel must be cooled, even after shutdown, because the radioactivity of the used fuel continues to provide heat – enough heat to drive the fuel temperature up to the melting point unless adequately cooled. We also understand that radioactivity cannot be shut off. We are therefore concerned that in case of a station blackout, there will be no power to drive the pumps, so there would likely be overheating of the fuel and over pressurization of the primary cooling system. We understand that failure to relieve the over pressurization fast enough could cause bursting of pipes in the primary cooling system –

which would complicate the cooling situation dramatically, even if power is restored. If those burst pipes happen to be among the thousands of aging pipes in the old steam generators (that were not replaced during refurbishment), then radioactive poisons could be released from the damaged fuel and might travel through the piping system to the crippled steam generator and escape out into the surrounding atmosphere without any filtering or containment.

In Appendix A of the briefing note, we read:

“We now have substantive reasons to conclude that the present, as-built, overpressure protection system relief valves do not meet the long-stated design requirements and do not meet critical ASME and CSA code requirements (by their size, location, isolation, number and testing) at any operating CANDU unit. These findings should compel immediate regulatory intervention. The system, if left as is, can have an adverse impact on public safety and utilities must also understand that their economic interests are equally challenged.”

Therefore, PRGI questions why this issue was not brought up in the 2022 ROR, nor to us as E-doc #6957534 (the Tracker) was last updated more than 14 months after receiving the briefing from the Working Group. Based on the Working Group’s briefing note, it seems as though there is consensus on the inadequacy of the pressure relief valves and that there are indeed safety concerns. We reiterate our concern that PLNGS is still operating while there is a serious unresolved safety issue.

It seems that failure to replace these inadequate degasser condenser valves (overpressure relief valves) with properly sized valves could, under adverse circumstances, prove devastating not only to the reactor and to NB Power but to all our relations, including the surrounding populations in New Brunswick, Nova Scotia, Maine, and possibly even farther afield.

The PRGI requests the findings of the Working Group be made public, considering the CNSC responsibility of ensuring that the health and safety of Canadians and the environment are protected, and their commitment to provide objective scientific information is disseminated to the public**.** Further we request an update on the status of the “assessment” discussed during the 2022 hearing and a notification of when the CNSC expects their report, in full, shall be shared. Let it be noted that we are both disappointed in the lack of timely information on an issue that we have expressed is of major concern to us, and that we are very concerned that the decision to not replace the degasser condenser valves to date, is based on economic and political concerns, rather than the prioritization of safety.

Concern #9 Financial Guarantee

In 2022 during the hearing for relicensing of PLNGS, as well as in our response commentary to the 2021 ROR and again now, we raise concerns about the inadequacy of the financial guarantee for decommissioning Point Lepreau and recommend that the licence be revised by CNSC to take into account this deficiency. In E-doc #6957534 (Tracker) the CNSC response is that it will be “willing to provide more information … if the Peskotomuhkati Nation is interested.” We are interested and have stated such to the CNSC at least twice in 2022.

At present, the legacy of decommissioning waste (which presumably will include the refurbishment waste as well as the operational low and intermediate level waste) is threatening because the waste has nowhere to go. We see no evidence that the CNSC staff is encouraging or requiring the licensee to find a site where this bulky long-lived radiotoxic waste can be safely isolated from the environment of living things for many thousands of years.

Such a search is both costly and lengthy. CNSC has a responsibility, as a self-described “life-cycle regulator” (Concern #3, CNSC response, E-doc #6957534), to ensure that the necessary funds and incentive are there for NB Power to take the time and spend the money to find an acceptable home for these wastes over the very long term. Under existing legislation, it appears that such radioactive wastes are a provincial responsibility and not the responsibility of the federal government. The legislated mandate of the Nuclear Waste Management Organization (NWMO) is limited to dealing with used nuclear fuel.

According to the United Nations Declaration on the Rights of Indigenous Peoples, there should be no storage or disposal of toxic wastes on Indigenous territory without the free, prior, and informed consent of the Indigenous community. This right has already been unjustly violated at the Point Lepreau site. As we made clear earlier in this document, we do not consent to having radioactive waste on our territory. In addition, given that the costs of dealing with these wastes is a provincial responsibility, the current arrangement requires that Peskotomuhkati citizens living in New Brunswick and all New Brunswickers will be burdened with paying to manage these wastes for millennia, making the injustice permanent.

New Observations

In reference to 1.4.4 Compliance verification program

With regard to s 1.4.4 we request a further information from CNSC staff regarding any additional reactive compliance verification activities for PLNGS (which were described within the ROR as compliance activities which relate to 'known or potential licensee challenges'). We are also interested in learning about any compliance verification activities related to PLNGS that indicate 'negative trends over time and/or deviations' from CNSC expectations.

In reference to Section 1.4.5 Safety assessment ratings

At s 1.4.5 on page 11 the 2022 ROR discusses, "For the Bruce A and B, Darlington, and Pickering sites, the NPP (nuclear power plant) and WMF (waste management facility) are assessed separately because they are regulated under separate licences and have facility-specific licensing bases. The WMFs at Point Lepreau and Gentilly-2 are governed by the NPP licences and are subject to the same regulatory requirements, so they are assessed together with their respective NPPs (as was done in previous regulatory oversight reports)." We seek further information on this matter to aid in our understanding of any implications of the difference in the two types of licenses, as well as the reason the NPPs are licensed differently.

In reference to section 2.2 Human performance management

At s 2.2 on page 14, the 3rd paragraph discusses, "Licensees reported 2 MSC violation at the DNGS, two violations at PNGS, three violations at BNGS A and B, and 2 violations at PLNGS that happened during 2022. All violations were of a short duration and the licensees took appropriate actions, e.g., calling in relief staff, holding over staff already present and operating in quiet mode.” We ask for a definition of ‘quiet mode’.

In reference to 2.3 Operating performance

At the bottom of page 15, in s 2.3 Operating performance, the ROR states, “unplanned transients indicate problems within a plant, and place strain on its systems.” It is further noted that the World Association of Nuclear Operators has a target of 1.5 trips per 7,000 hours critical, which PLNGS did not meet in 2022. On page 119 details regarding the two 2022 Point Lepreau reactor unplanned reactor trips are provided. Each of these unplanned fast shutdowns required the use of one or more of the two fast reactor safety shutdown systems that are available in every CANDU reactor (except for the two Pickering “A” reactors).

The first unplanned trip occurred on August 2, 2022, and required the use of Shutdown System 1 (SS1), i.e. the sudden insertion of spring-loaded neutron-absorbing shutoff rods into the core of the reactor to stop the chain reaction very quickly. The second unplanned trip occurred on December 14, 2022, and required both shutdown systems SS1 and SS2. Shutdown system 2 uses a liquid neutron absorber, gadolinium nitrate, that is rapidly injected into the moderator through nozzles in the calandria vessel, stopping the chain reaction by absorbing the neutrons.

These two unplanned trips took place over a period of 134 days, for a frequency of about 6.2 x 10-4 = 0.00062. That frequency is almost four and a half times higher than the target frequency for unplanned trips, set at 1 per 7000 hours, or 1/7000 = 1.4 x 10-4 = 0.00014.

Both of these unplanned trips were caused by electrical failures. In the first case, an overcurrent to ground fault shorted out the Heat Transport pump motor, leading to unacceptably low flow of coolant in the fuel channels, triggering an emergency fast shutdown. In the second case, an electrical fault on a cable from the Unit Service Transformer caused a partial loss of power from the grid. This caused a reactor setback from the automatic Reactor Regulating System, automatically triggering the two, fast shut-down systems to spring into operation.

It is concerning that two such important electrical failures would happen in such a short time frame. As discussed above in Concern #8, we are concerned about the possibility and associated consequences of boilers running dry because of a lack of power to pump the water through them. The data from Point Lepreau in the 2022 ROR indicates that the probability of such an event is much higher – 4 or 5 times higher perhaps – than is assumed in the CANDU safety analyses that are performed from time to time.

We reiterate our position that it is unacceptable that there has been a 16-year delay in ordering the replacement of all the inadequate overpressure relief valves in CANDU reactors with fully adequate pressure relief valves. Such major accidents are indeed unlikely, but they do happen, and the consequences can be staggering.

In reference to 2.12 Security

In s 2.12 on p31, the 5th paragraph refers to a “schedule of deferred security exercises”. PRGI requests more information regarding this schedule and its reason for existence.

In reference to 2.13 Safeguards and non-proliferation

Under table 10, the 2022 ROR states, “The IAEA considered most of the inspection results to be satisfactory.” PRG requests to know about the results found to be unsatisfactory.

In reference to 2.15 Indigenous Consultation and Engagement

We are pleased to see the addition of both appendices E and G, and encourage the CNSC, as has been discussed with the CNSC in the past, to support a forum for nuclearized Indigenous Nations to discuss common issues and concerns. PRGI would be pleased to co-develop such an initiative.

On page 36 the ROR references the participation of the PRGI in environmental monitoring efforts for the Point Lepreau site, however, in section 3.7.9 (Environmental Protection) fails to include any mention of how our participation factored into the CNSC's ranking of "adequate provision" having been made by NB Power for the protection of the environment and public health. As the CNSC's Indigenous Knowledge Policy states, 'the CNSC acknowledges the importance of...considering and reflecting Indigenous Knowledge in its assessments and regulatory processes.' PRGI submits the ROR is precisely that - a regulatory and assessment process - and yet there is no reflection or acknowledgement that any Indigenous Knowledge from PRGI informed the conclusions reached. We do acknowledge that Appendix E specifically reflects how IK is being integrated in certain circumstances and suggest the next step is to also integrate and reflect this knowledge, in the appropriate section of the report. We therefore continue to recommend the CNSC work to ensure its technical framing and ranking of licensee activity be assessed in tandem with CNSC policy which purportedly seeks to include IK. Both the CNSC's Indigenous Knowledge Policy and PRGI aim for a reflection of learning based on meetings and engagement.

In reference to 2.16.1 Public information and disclosure programs

On the second paragraph of page 41, we note that OPG has initiated a self-assessment to identify areas of improvement related to a global name change which will be completed Q4 2023. We request information on the outcome of this assessment as we also object to the OPG rebrand of Waste Management Facilities to 'Nuclear Sustainability Services', as the initiative could be precedent-setting across the Canadian Nuclear industry. We note that the name change was opposed by not only intervenors but also the Commission itself.

In reference to 2.16.3 Financial guarantees

Please refer to our previous comment under 'Concern 9 – Financial guarantees.

In reference to 2.16.5 Forum between the CNSC and Canadian ENGOs

This seems to be a new and appreciated section of the 2022 ROR.

In reference to 3.7 Point Lepreau Nuclear Generating Station

We note that the introductory discussion (3.7.0) should expand to include discussion of intermediate and low-level waste and be transparent in the fact that some of this waste travels internationally to meet waste minimization goals of PLNGS.

Under the subheading of 'Event Initial Reports' we introduce further discussion on the event report relevant to December 14, 2022. The first detailed event report for the December

2022 incident was dated April 21. Note that at a meeting with CNSC on May 11, PRGI raised questions about the lack of detail in the April 21 report. A second detailed event report with more details was submitted, dated May 26, however we continue to have questions that are not answered in the second report. We are providing information and asking questions about this event so that CNSC staff and Commissioners appreciate the challenges we face trying to decipher important documentation.

Partial context for our concern is the high levels of tritium released into the environment by PLNGS referenced elsewhere in this document. We note that CANDU reactors are, compared to all reactors globally, among the highest emitters of tritium per MWh, and that the Point Lepreau reactor is the highest emitter of any CANDU reactor, which likely makes PLNGS the highest emitter of tritium in the world.

In this context, the volume of tritiated water released in the December 2022 incident is a core metric that should be included in the event report. However, this information is missing in both versions of the report. Again, we point out that the scope of many CNSC documents makes fulsome understanding of consequence illusive.

Both event reports indicate a release of about 1.75 litres per second without specifying how long the leak lasted. How long did it last? The reports are unclear. The CNSC response report (CMD23-M7) dated January 13 suggests 13.45 hours, but the NB Power report says the event lasted for ~66 hours.

We request answers to the following outstanding questions about the tritiated coolant that escaped during the event:

- If the leak lasted 13.45 hours (the lower figure), at least 84,735 litres of tritiated coolant had spilled out, more than an average swimming pool. About the event itself, we ask - is this the amount of tritiated water released? About reporting, we recommend quantitative information be included in future reports.
- How is it possible that "the inventory [of heavy water] was collected and safely returned to the system"? Wouldn't the heavy water have to be thoroughly cleaned before returning to the system? We recommend that in future event reports, more detailed steps to rectify the issue are documented.
- As rightsholders with an established relationship with PLNGS, we understand that we can call PLNGS representatives for further information on such issues. However, we also are concerned that stakeholders understand how to access information on nuclear-related incidents within our homeland. We recommend it is clear to those accessing an event report the process to formally request additional information.
- In April 2023, a heavy water leak was reported at the Bruce Power station.[9] We seek further clarity on the similarities and differences between the coolant leaks in the two different CANDU plants and recommend that both rightsholders and the public need a clear and easy process to access such information.

Concluding Remarks

PRGI seeks fulsome discussion with the CNSC on each of the unresolved items we have provided comments and recommendations on, in this document. However we also note, we feel under pressure to 'skip ahead' to concentrate our energies on the ARC-100 SMNR License to Prepare Site application, and associated consultation.

In making these submissions, we underscore that the CNSC and nuclear proponents must understand that colonial processes – from which their legitimacy is derived – has destroyed much our capacity in the first place, and we now feel a demand upon us to repair the damage to our capacity instantly. Even with the capacity funds the CNSC has recently made available to

[9] https://london.ctvnews.ca/bruce-power-reports-heavy-water-leakage-1.6373769

PRGI (and we are grateful to have), finances alone don't immediately enable our ability to respond to the multitude of issues, while hiring the people with the skill sets necessary take on the nuclear portfolio (while at the same time trying to build the community capacity, by hiring Nation members), while negotiating financial support and agreements, while submitting meaningful responses to the ROR and concurrently submitting meaningful responses to the Government of New Brunswick 's draft guidelines for the ARC-100 EIA (these last two deadlines were only 2 days apart).

In our most recent meetings with the CNSC staff, while discussing a potential long-term relationship agreement – which we feel is the proper first step, we are reminded of the urgency associated with the potential ARC-100 consultation agreement. We therefore continue to return to our priority recommendation and position – namely that consent was never sought, nor granted from the Peskotomuhkati, for the development of the PLNGS on the shores of the Bay of Fundy and the responsibility now rests with the CNSC to shift its laws and policies to recognize and respect our rights to self-determination and inherent self-government.

As mentioned at the beginning of this document – though we have received responses to our 2022 recommendations, we seek to assist in making CNSC documents more understandable to both rightholders and stakeholder, as well as to encourage contextual discussion. Therefore, although we have been advised that it is not within the scope of the ROR to do so, the inclusion of 10-year trends, and discussion of root causes, would help reveal the depth of any problems, and may assist in defining solutions which are responsive to concerns raised.

Despite our thorough review of the ROR, the depth of our comments is indicative of the expansive gaps in the ROR, which like other RORs and CNSC documents, unfortunately lacks context for those interested in understanding whether or not the health and safety of persons and the environment is indeed being protected from nuclear-related risk. We continue to press the CNSC Staff to provide information related to the reasons for the various CNSC regulations – i.e., to protect the environment and us, from the many harmful biological effects of chronic or acute exposures to radioactive materials, and the multitudinous pathways of radionuclides through the environment and through the body.

Appendix A – Edoc 6768776-v8-Briefing_Note_on_Degasser_Condenser

BRIEFING NOTE TO Mike Rinker DAA DG

Briefing Note on Degasser Condenser Relief Capacity for CANDU Reactors

June 17, 2022

ISSUE OR OBJECTIVE

Do the overpressure protection devices installed on the degasser condensers of operating CANDU reactors have adequate discharge capacity, location, size, and number? Sufficient discharge capacity is required to limit overpressure when a mismatch between effective thermal load into the primary fluid power and heat removal capacity occurs during postulated accidents. This briefing note aims at providing a technical basis to answer the above question.

BACKGROUND

This question is not new, as at least twice in the past, intervenors had raised the issue, and the matter was closed. The industry and the CNSC conducted two previous reviews of the relief valve capacity [1, 2, 3]. Several papers have continued to raise the issue in conference proceedings. These papers disclosed new evidence overlooked in earlier investigations. One example is using a CFD analysis to justify air tests to obtain a steam discharge. The study performed no mesh density independence analysis [3]. The investigation provided neither a rigorous validation of the model as per the requirement [4] nor an uncertainty analysis using CFD best practice guidelines [5]. An independent review of the disposition of the intervenor's issues was conducted [6]. This study referred to another study conducted by an independent ASME Code specialists who concluded that the licensees are not required to conform to the overpressure relief requirements of ASME Boiler and Pressure Vessel Code Section III during a BDBA [7]. In that report, the ASME specialist stated, "This evaluation did not consider the points made in the documents with reference to the options used in the design of the overpressure protection system, the accuracy and completeness of the calculations and tests and whether or not the existing systems are adequate." A working group within the Directorate of Assessment and Analysis started to review the latest evidence as part of an email exchange between the President and the author of the conference papers. The author of the conference papers, who was hired as a consultant, provided additional analysis in Appendix A.

<u>What accidents are likely to cause overpressure?</u>

The overpressure of the Primary Heat Transport System may occur due to a class of accidents linked to a loss of boilers as heat sinks. Examples leading to loss of heat sink are:

- Loss of inlet feedwater flow to one or more steam generators
- Pipe break in a steam line
- Pump failure and pump seizure
- Loss of Class IV power
- Loss of Class IV power resulting in a partial drain of primary inventory into the below header pressurizers at Darlington and Bruce
- Primary system LOCA with and without reactor trip

AECL believes that sustained loss of heat sink belongs to Beyond Design Basis Accidents [2], and therefore the current design requirements are not required to be met for an existing old design. However, for a new design such as the ACR-700 design, the relief valves open when the pressure in the bleed condenser reaches the value of 12.27 MPa(g), with a relieving capacity on each valve bounded by the sustain loss of all heat sinks event, to give a relief capacity of 21.3 kg/s for steam [8]. Given the importance of a robust system to public safety, a design change for the component as in ACR-700 is justifiable.

Systems available to limit overpressure

There are two systems available to limit the Primary Heat Transport System (PHTS) pressure. The first system is the two shutdown systems acting alone can terminate neutron chain reaction and prevent failure of the PHTS due to overpressure, provided the heat sink capabilities are fully poised and available to remove the decay heat. The second system consists of a pressurizer, a train of valves such as the liquid relief valves and the safety relief valves on the degasser condenser. The latter system is the only pressure relief system when the heat sink function of the reactor is lost.

Design Requirements for pressure relief

There are three design requirements for pressure relief.

1. The design requirement of the pressure relief valves is to protect the PHTS by relieving the coolant volume swells during various disturbances in power to arrest pressure escalation in the PHTS. The nuclear class in CSA Standard CAN3-N285.0-95, Clause 7.4.2 and Section III of the ASME Boiler and Pressure Vessel Code BPVC III 1 NB-2021 provide the overpressure protection design requirements. From the Qinshan Overpower Protection Report [9] the requirements are that under all operating conditions, the relief valves must open when the pressure in the degasser condenser reaches the value of 10.16 MPa(a), irrespective of the pressure signal to the actuator. The relieving capacity of each valve is 26.66 kg/s at 268°C of liquid. The sizing of the degasser condenser relief valves is based on the premise that the PHT System is experiencing a dual accident: loss of inventory control and loss of heat sink. The reactor outlet header is at 110% of the PHT relief valve set pressure, i.e., at 11.37 MPa(a), and the degasser condenser is at 110% of the relief valve set pressure, i.e., at 11.17 MPa(a).
2. The pressure relief valves shall be close to the significant source of overpressure anticipated to arise within the system under the conditions summarized in the Overpressure Protection Report (NB-7200).
3. The back pressure that may exist or develop shall not reduce the relieving capacity of the relieving device(s) below the level required to protect the system.

How have other jurisdictions designed pressure relief?

The Surry Nuclear Power Station located in southeastern Virginia is a 2,587 MW_{th} PWR, an equivalent thermal power to a typical CANDU at Bruce or Darlington, has three safety relief valves and 2 Pilot Operated Relief Valves with a steam relief capacity of 112 kg/s and 53 kg/s, respectively. At Georgia Power's Vogtle (3,625.6 MW_{th}), the three safety relief valves can discharge at 17 MPa of 159 kg/s with an additional 53 kg/s from the 2 PORVs. The single valve liquid relief capacity satisfies pressurization challenges due to liquid swell at full power. These plants have redundancy and a testing program to assure functionality. The Design Requirements for PWR reactors state that they should have sufficient capacity to preclude actuation of safety valves during normal operational transients, when the reactor is operating at the licensed core thermal power level [10]. They also require that the valves should be tested and inspected.

<u>How well does CANDU pressure relief system meet the requirement?</u>

1. The safety relief valves downstream of the Degasser Condenser have a specified steam relief capacity of 4 kg/s. When tested, the actual relief capacity was lower than the specified relief capacity. For example, when tested in 2001 by OPG at Wylie Laboratories under the manufacturer Bopp & Reuther's supervision, the Pickering relief valve's combined steam relief capacity was 120 g/s for its two relief valves [11, 12]. The power of a single Darlington unit is 2776 MW_{th}, and the Bruce unit is 2832 MW_{th}, which is not too far from the Surry Nuclear Power plant. For Qinshan, the specified relieving capacity is of each valve is 26.66 kg/s at 268°C and 10.16 MPa(a).
2. The pressure relief valves in the operating stations in Canada are not close to the source of overpressure expected. As Figure 1 shows, they are several meters away from the headers.

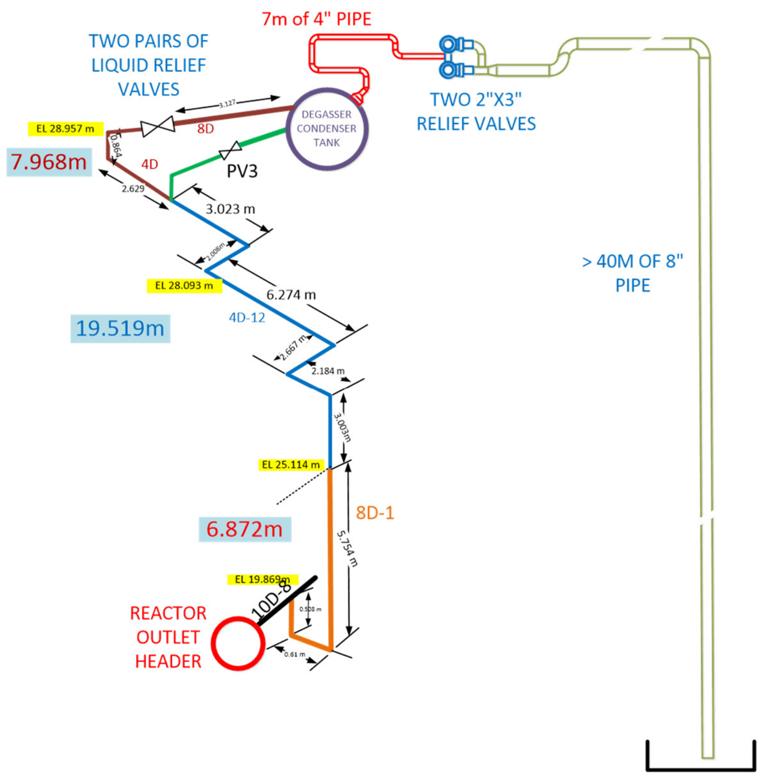

Figure 1: Distances Between Header, the Liquid Relief Valves, the Degasser Condenser Safety Relief Valves and the Downstream Piping to Collection Tanks

3. The potential backpressure from the >40 m long piping downstream of the relief valves will significantly increase pressure. The pressure required to open the valves will exceed the pressure needed for the design.

CONCLUSION

After significant debate and analysis, the Working Group concludes that the relief capacity of the degasser condenser is inadequate for beyond design basis accidents such as a prolonged loss of all ac powers leading to a sustained loss of all heat sinks. It does not meet the current ASME and CSA's stated design requirements. A comparison with a similar Light Water Reactor system also points to the inadequacy for beyond design basis accidents. Post-Fukushima enhancements, including Emergency Mitigating Equipment and Containment Filtered Venting System, which have been added in Canadian nuclear power plants, provide countermeasures to stop the progression of severe accidents and minimize their radiological consequences to the public.

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Approved by: name and title, no signature

Date: 2022-06-17

Appendix A

MANAGEMENT BRIEFING NOTE ON CANDU PRIMARY COOLING SYSTEM OVER-PRESSURE PROTECTION SYSTEM DESIGN REVIEW FINDINGS

Dr. S. Nijhawan

SUMMARY

The working group performed an extensive design assessment of the pressure and inventory control system of CANDU primary heat transport system because concerns had been repeatedly raised by an intervenor about its design inadequacy implying serious implications on reactor safety. CNSC President Velshi directed us to engage with the intervenor on a number of other issues he raised as well and come to a common understanding.

Reactor pressure boundary integrity is assured in reactors by action of inventory control, pressure relief and reactor regulating systems. However, it is only the primary heat transport system's pressure relief system whose assured adequacy is critical to HTS pressure boundary integrity in most scenarios. An early pressure boundary rupture in any event that requires action of overpressure protection can have cascading and disproportionate consequences. The focus of our investigation is the location, size, number and relief capacity of relief valves that open into containment to mitigate any sustained overpressure and compliance with applicable codes and design requirements. These are the Degasser Condenser relief valves (DCRVs).

We examined process and safety requirements & applicable code recommendations for overpressure limits; reviewed design documentation, as-built geometries, LRV failure reports, DCRV test data and past licensee submissions on the issue. We also examined previous intervenor submissions, published papers and past CNSC documents on DCRVs. Since the design basis and safety objectives must be pretty much the same for overpressure protection in PWRs and PHWRs, we also obtained design information on similar systems for other reactor designs and examined their actual implementations, analyses in support of code compliance and pertinence to CANDUs. We examined the relevant historical CANDU design documents and noted relative uniformity across various CANDU stations in design over decades. We also setup models that demonstrated that just the location of DCRVs would never allow full compliance with codes and that relief valve upgrades following the 1994 Pickering LOCA event that resulted in DCRV replacements of the same liquid discharge capacity but smaller sizes, further reduced the effectiveness of the system by increasing the already high flow resistance caused by long pathways between headers and relief valves.

We now have substantive reasons to conclude that the present, as-built, overpressure protection system relief valves do not meet the long-stated design requirements and do not meet critical ASME and CSA code requirements (by their size, location, isolation, number, and testing) at any operating CANDU unit. These findings should compel immediate regulatory intervention. The system, if left as is, can have an adverse impact on public safety and utilities must also understand that their economic interests are equally challenged.

BACKGROUND

Pressure relief valves at all pressurized-water power reactors of all designs are required for the same reasons and must fulfill the same functions with the same expectations of sufficiency and reliability for a similar spectrum of transients and accidents. ASME Boiler & Pressure Vessel Code Section III NB7000 summarizes requirements for maximum overpressure to 10% over design pressure for 'anticipated events' and 20% for 'unanticipated events.

Overpressure relief design solutions are relatively simple and would in principle be similar in capacity and effectiveness between PWRs and PHWRs of similar thermal power. To cover all mismatch between energy generation and removal, excess energy must be specified to be removed by an energy equivalent steam discharge or discharge of, volumetrically equal to steam generation, liquid swell to cover all scenarios. The latter is critically important for horizontal channel PHWRs in certain SBO scenarios that cause a breakdown of thermos-syphoning and local generation of steam.

In a typical PWRs there are 3 to 5 direct, fast acting and power operated relief valves directly on pressurizer with combined steam relief capacity 100 to 250 kg/s. In CANDUs the overall steam relief capacity is from 1 to 3 orders of magnitude smaller. A simplistic equivalence of 1 kg/s steam discharge to 1 MW of excess thermal power at 10 MPa quickly points to the inadequacy of CANDU relief valve capacity for long periods of time after reactor trip. There is nothing in PHWR designs that justifies the difference.

There are two '50%' relief valves located not near the pressure sources at headers or pressurizer but downstream of other valves (LRVs), a vessel (Degasser Condenser) and ~40m long piping between the pressure source and relief. The back pressure caused by the pressure drop by the required relief rate of steam would cause a backpressure of the order of 10% overpressure allowed by code. The '100%' capacity is strangely defined based on liquid swell rate at shutdown conditions and a simple volumetrically equal steam relief capacity for two valves is as low as 4 kg/s in CANDU-6s. Actual tested relief capacities are even lower and confirmed by multiple tests to be a mere 120 g/s for 2 valves at Pickering.

Overpressure relief capacity must be certified by tests and many regulators (e.g., NRC) require periodic testing of actual relief valves. No verbal arguments or 'modelling' is allowed to substitute these requirements except to prorate the certified design values to higher pressures under certain conditions. Examination of test reports and correspondence reveals problems encountered in mere bench testing of CANDU relief valves by BP and OPG and supports the Code requirements of low scatter and repeatability of claimed relief capacity. The industry response over the 2 decades that this issue has been open, is erroneous because their claims of DCRV steam relief capacity exceeded by a large factor even what would be theoretically possible.

CANDUs at Wolsong, Bruce and Pickering had a number of LRV/RV failures causing LOCAs and there is no relief valve testing program in place. Testing of any safety critical relief valves is an important acceptable practice in nuclear and chemical industries. Past issues with PWR relief valve reliability (recall stuck open relief valve at TMI) have been addressed by NRC mandated periodic testing after TMI. It is only in the CANDU primary circuit overpressure protection systems that the long relief path is also indirect; ultimate steam relief capacities are low, and the reactor design allows no redundancy or sharing of boiler heat sinks across the core (multi loop PWR designs where boilers share whole core thermal load vs. a segmented heat sink design where a group of fuel channels are served by only one dedicated boiler in most PHWRs).

It was first in assessments for a SBO that the relief capacity of DCRVs was shown to be inadequate, but CANDU safety reports had already documented the inadequacy indirectly a number of times where overpressure exceeding code requirements were noted and brushed off inexplicably as still being lower than rupture pressures. A predicted (and documented in safety report) close to 100% overpressure from a postulated sudden loss of inventory from one boiler (Pickering), of a postulated break in feedwater pipe at SG inlet is a good example. This is not a low probability event. Feedwater pipe thinning and failures have multiple precedents in PWRs arising from carbon steel pipe thinning, with some resulting in multiple fatalities.

One quickly realizes that there is nothing in a PHWR concept that requires the overpressure protection system to be so different and so irresponsible in so many ways. We have also confirmed PHWR overpressure design inability to meet even the stated, albeit erroneous, primary design basis of liquid relief. We have summarized the multiple code requirements that are violated and found no justification for the current design contravening established engineering practices.

Overpressure design deficiency is not a new issue. First raised in 1997 at AECL and in 2001 with CNSC, it has been the subject of numerous meetings & technical papers. It has been raised since 2014 at all relicensing hearings for all stations in Canada. It is only now that the staff have been asked to look at it in a comprehensive, multi-disciplinary team approach with direct intervenor engagement. It has become clear that the major initial 1990s findings and subsequent confirmation by tests of critically low relief valve steam relief capacity is but one of many design features of concern. It is also clear that the industry submissions on the topic of DCRV steam relief capacity were baseless and should have been better investigated. Other design deficiencies include, relief valve location, testing and lack of redundancy. Optimal solution will require will and industry wide engagements and resolution of this issue must take priority. This generic CANDU design inadequacy has safety implications both for design basis accidents that the reactors are licensed to meet and for beyond design basis accidents for which measures must now, post Fukushima, be demonstrably undertaken to ensure that risk to public can be minimised by engineered means.

We have further details documented on how and why stated design requirements and applicable ASME code recommendations are not met and how worldwide engineering practices in reactor main heat transport system pressure boundary overpressure protection, are contravened.

APPENDIX A

DESIGN BASIS FOR OVERPRESSURE PROTECTION SYSTEM

Dr. Sunil Nijhawan

Just as is the case for all operating pressurized water reactors, the design basis for CANDU PHWR Primary Heat Transport System over-pressure protection system includes a class of accidents leading to inability for boilers to be adequate heat sinks. Examples of accidents so considered within the design basis include:

1. Loss of inlet feedwater flow to one or more steam generators
2. Loss of steam generator inventory due to inlet pipe rupture
3. Pipe break in a steam line
4. Pump failure and pump seizure
5. Loss of Class IV power
6. Loss of Class IV power resulting in partial drain of primary inventory into the below header pressurizers at Darlington and Bruce
7. Primary system LOCA with and without reactor trip

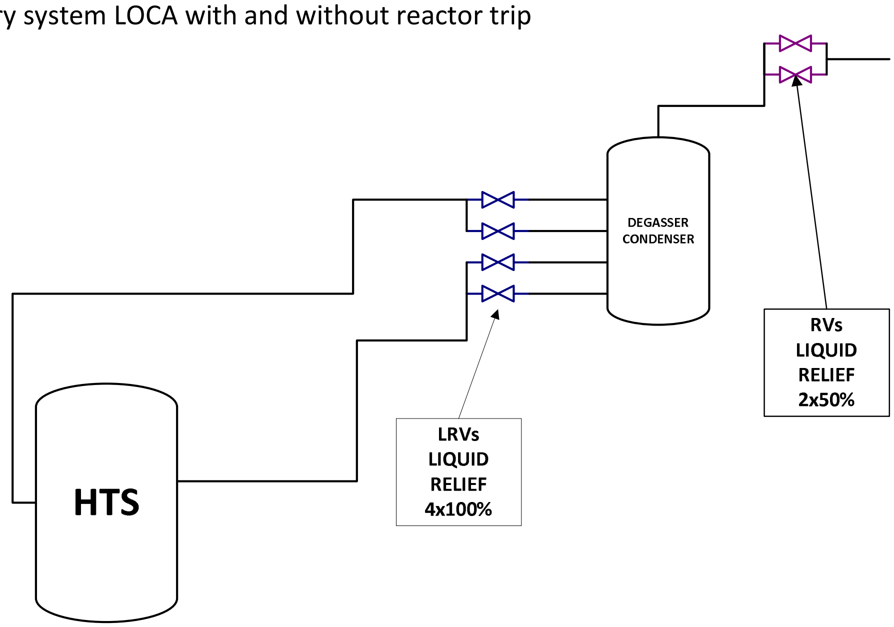


Figure 1: Typical PHWR CANDU PHTS pressure relief – indirect and away from PHTS; discharge path interrupted by other valves, vessels & systems.

Table 1:

	Bruce A (2013)		Bruce B (2013)		Darlington (2011)		Pickering A Units 1&4 (2011)		Pickering B Units 5-8 (2011)		CANDU-6 PLGS (2011)	
BCRV/DCRV Equipment Number	33320-RV17	33320-RV18	33320-RV17	33320-RV18	63332-RV25	63332-RV26	33320-RV108	33610-RV5	33320-RV163	33320-RV108	3332 -RV11	3332 -RV21
Nominal Valve Size (inches)	2x3	2x3	2x3	2x3	4x6	4x6	2x3	3x6	2x3	4x6	2x3	2x3
Max Lift (mm)	5.5	5.5	5.5	5.5	4.15	4.15	7.0	6.1	4	5.5	5	5
Opening Set point (MPa)	8.3		9.43		RV25=10.17 RV26=10.681		RV5=8.6 RV108=8.1		8.6		10.09	

A schematic of the CANDU-6 o*verpress*ure relief system in Figure 1. In all CANDU design documents, LRVs are described as valves protecting HTS from over pressure, and the downstream Relief Valves (RVs) described as providing overpressure relief services to the degasser condenser. Practically speaking, the relief valves mounted ~7m downstream of the degasser condenser (Figure 3) are de facto PHTS relief valves and all discussion about relief capacity in this technical note is about these relief valves. LRV's are not relief valves in terms of their opening passively, proportional to overpressure. They are kept closed by pneumatics and open on pressure signals using a 48V actuator. They are adjusted to open a certain size to instill a certain pressure drop (25 psi in CANDU 6) at a certain subcooled water discharge (26 kg/s) through them. Note that even the LRV design basis is a reactor shutdown state (According to a CANDU 6 Overpressure protection report -Reference [i], section 12.4.1.2– "*The heat transport relief valve capacity was determined by considering postulated process failures which lead to pressurization of the Heat Transport System when the reactor is not at power"*) and the downstream relief valves are also sized for a shutdown state and <u>even</u> a liquid swell relief capacity (0.036 m^3/s) of the relief valves is much smaller than required by the stated design basis (Figure 2) to even remove liquid swell in a subcooled PHTS from decay heat.

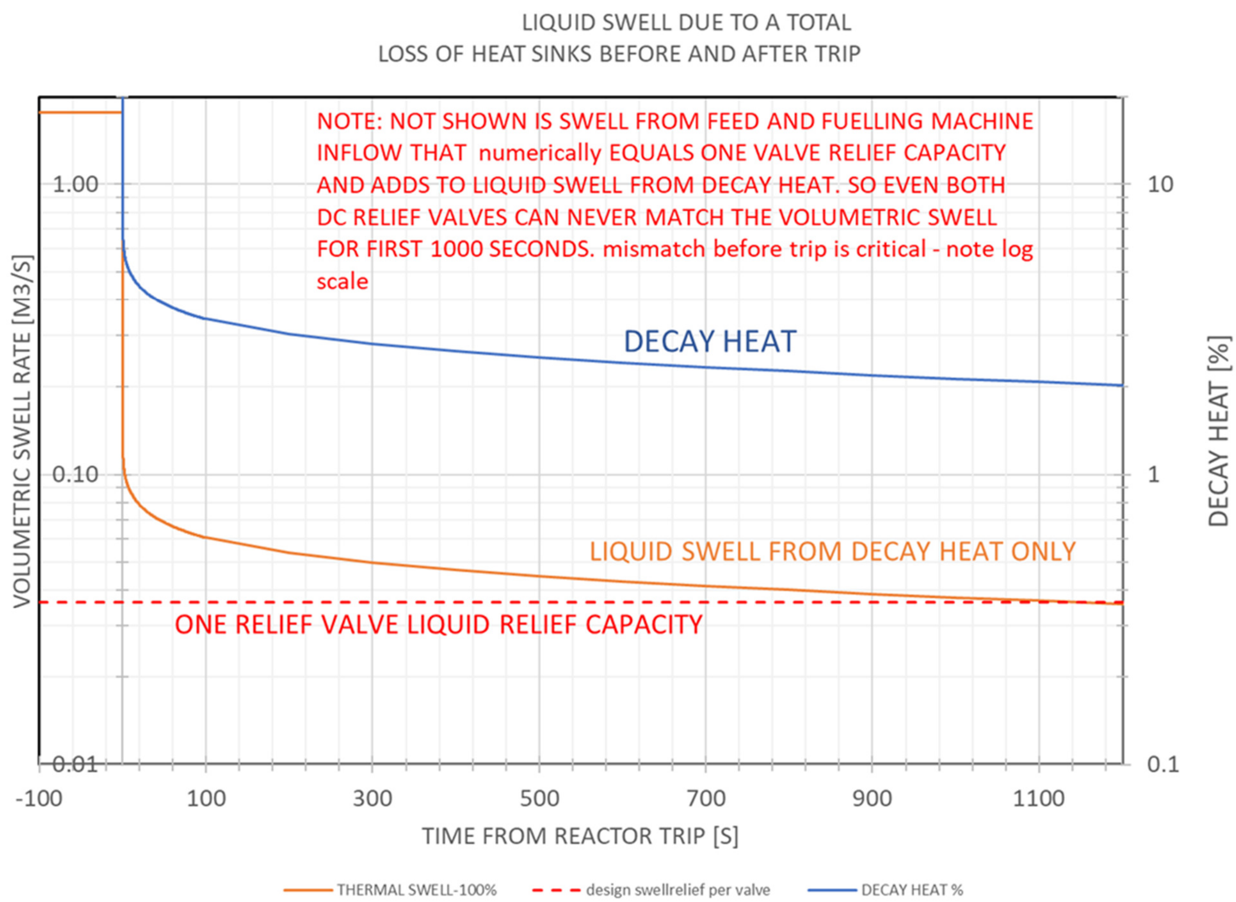


Figure 2:Even liquid swell rate, the erroneous design basis is higher than valve relief capacity.

No other reactor overpressure relief valves specifications are based on a liquid relief capacity. Most have relief capacities based on 2 orders of magnitude higher steam relief capacity than has been for CANDU reactors.

BEYOND DESIGN BASIS ACCIDENT OVERPRESSURE PROTECTION SYSTEM

Consideration of reactor response to a sustained loss of all AC power – a Station Blackout (SBO) is now within the realm of serious consideration, especially after TMI and Fukushima. Thermal hydraulic assessments include a depletion of boiler secondary side inventory, and a fundamental expectation is that it does not cause an uncontrolled overpressure and a rupture of pressure boundary. In CANDUs this cannot be avoided although all analyses undertaken so far don’t even touch the subject of comparing relief capacity against requirements. They do not model the PHTS dynamics and just assume that the relief valves designed and tested for 2 kg/s of steam can magically relieve the required (>30 kg/s) and cause no overpressure. So how much relief capacity do we need? Simply speaking a steam relief capacity in kg/s equal to one MW decay heat at 10 MPa and 30 kg/s of steam relief capacity is a rough target that CANDUs fail to meet by an order of magnitude or more. Liquid swell caused by vapour generation in stagnant fuel channels would require very large volumetric 2 phase relief capacity. That will be amply covered by a steam relief valve system similar to the ones in Westinghouse PWRs.

VALVE FLUID DISCHARGE CAPACITY

As a basic requirement, the Relief Valve fluid discharge capacities must satisfy excess thermal loads that would otherwise be removed by engineered means rendered incapable or deficient by an accident or a transient.

All operating pressurized water reactors will discharge either a mixture of liquid water and steam or pure steam through relief valves on an overpressure and all PWRs support a steam relief capacity of more than 100 kg/s with the unspecified liquid relief capability coming in at about 500 kg/s in tests to be about 5 times more numerically than for steam at their pressures[1]. Only CANDU reactor designers singularly specify a liquid relief capacity of about 50 kg/s through 2 valves to compensate for a small liquid swell rate from a loss of heat sinks at decay power about 30 minutes after trip. Specified to be volumetrically equivalent to liquid discharge, the small discharge capacity of steam is a secondary specification giving a steam mass relief capacity of about 5 kg/s from 2 '50%' valves. Actual tested relief capacity for steam through 2 valves is as little as 120 g/s for Pickering A valves and 2-4 kg/s for Bruce and CANDU-6 relief valves.

Note the order of magnitude difference between PWR and PHWR designs. Actual implementation of pressure relief in CANDU reactors creates additional issues. The fluid removal through pressure relief out into the containment is through relief valves at the end of 40m long pipes long path lie additional power operated isolating valves (LRVs) and vessels (Degasser Condenser and Pressurizer). Specificity of CANDU reactor horizontal, separate fuel channel design requires both liquid and steam volumetric discharge rate capacities be large and equal for the two-spring loaded pressure relief valves downstream of the Degasser Condenser.

Just for illustration purposes, note that 1 kg/s of steam production is from about 1 MW of power at all pressures greater than 13 MPa (with about 20% more power at 10 MPa and 20% less at 16 MPa). Just for comparison, at Surry 2600 MWth PWR - a typical PWR of about same thermal power as typical CANDU at Bruce or Darlington- there are 3 safety relief valves and 2 Pilot Operated Relief Valves with a steam relief capacity of 112 kg/s and 53 kg/s, respectively (Reference [ii]), corresponding to about 4 times the decay heat at a time that the boilers could become ineffective heat sinks in a station blackout. At Georgia Power's Vogtle, Comanche Peak and many other PWRs - –References ([iii],[iv]) the 3 safety relief valves can discharge at 17 MPa a total of 159 kg/s steam with an additional 53 kg/s steam from the 2 PORVs (liquid relief capacities would be numerically many times more). Their single valve liquid relief capacity satisfies any liquid swell pressurization challenges at full power, although there are no scenarios where just a liquid relief would occur beyond a small initial period. Their sheer number assures redundancy, and a robust testing program (References [v], [vi], [vii]) assures functionality and multiple NRC/industry feedback channels on relief valve degradation.

In comparison, the two CANDU-6 safety relief valves, located downstream of DCRVs have a specified steam relief capacity of typically only ~4 kg/s with actual steam relief capacities tested much lower. For example, for Bruce valves, water, and steam tests in 2005 at Wylie Laboratories indicated an actual capacity between 1 and 3 kg/s. When tested in 2001 by OPG at Wylie under manufacturer Bopp &

[1] For example, Diablo Canyon PWR SRV Steam relief capacity = 53 kg/s per valve through three spring-loaded, enclosed poppet-type, self-actuated angle relief valves with backpressure compensation. Same for Comanchee, Watts bar. Surry PWR has five relief valves (3 SRVs - 37 kg/s steam each and 2 PORVs- 26.5 kg/s steam each).

Reuther supervision, the Pickering relief valve combined steam relief capacity was only 120 g/s for its 2 relief valves [References, [viii] [ix]], when a steam relief capacity over 200 times more would be required to mitigate consequences of a station blackout scenario not too different than that resulted at Fukushima; and postulated as well in a number of design basis accidents.

The present overpressure protection system has already caused a number of cases of loss of primary coolant at Wolsong and Bruce reactors due to spurious actuation and failure to reclose of LRVs. A rupture of pressure boundary involving a 1994 loss of coolant and actuation of the ECC at Pickering, was likely due to its inadequacy in both liquid/steam relief capacities after an apparent reactor HTS overpressure and expected chattering of relief valves located tens of meters from pressure source.

OVERVIEW OF OVERPRESSURE SYSTEM DESIGN FEATURES THAT ARE FAULTY

Designed to mitigate a pressurization due to a loss of heat sinks at all operational states or an undue injection of additional coolant, the present CANDU over-pressure relief system cannot perform its stated functions because of the following reasons:

1. **Relief capacity too small for both liquid and steam relief for a wide range of events**
The relief valves are not sized to provide adequate energy and volumetric relief capacity that exceeds the mass and energy inputs that may cause over pressurization. As a result, the Code specified overpressure tolerances for corresponding service levels are likely to be exceeded. The relief valves are erroneously sized for liquid relief only (Reference [x], [xi]) and not directly and primarily for steam relief, which for a pressurized water reactor is a fundamental design feature. In technical specifications for PWRs one does not specify the liquid relief capacity, one specifies steam relief capacity, whose volumetric equivalent in liquid is always found to be sufficient to cover any cases of liquid swell. Compared to a PWR, the design steam relief capacity in CANDU reactors is about 10 to 50 times too small. Even the liquid relief capacity is inadequate to mitigate an overpressure transient such as the one caused by a rupture of a feedwater pipe at the inlet to a steam generator at power. Such pipe failures have occurred at a number of nuclear power plants due to steel pipe thinning and have even resulted in worker deaths in Japan and the U.S. An unacceptable overpressure by up to 100% following such an event is already documented in some licensee safety reports (e.g., Pickering SAR – Appendix 7, Figure 7-17) while the Code requirements limit such overpressure to 10%. It is shown by simple hand calculations and confirmed by tests that neither liquid, nor steam relief capacity is sufficient and an uncontrolled overpressure is inevitable.

2. **Relief valves located too far away from pressure source in a manner that a Code recommended and desirable for safety, unobstructed path from the reactor to the pressure relief valves is not made available**
The CANDU pressure relief valves (RVs) are typically located ~7m downstream of degasser-condenser vessels which are themselves located over 30m away from reactor headers with intermediary power operated valves (LRVs) near their inlet (Figure 3). As a result, the pressure-drop over these pipe distances of ~40m for the required relief rates for inlet saturated water or steam far exceeds the allowable magnitudes and likely to cause an undesirable overpressure in the primary heat transport system. The additional, long ~30m path from the relief valves to the basement sump presents additional back pressure likely to reduce the valve mass & energy relief capacity.

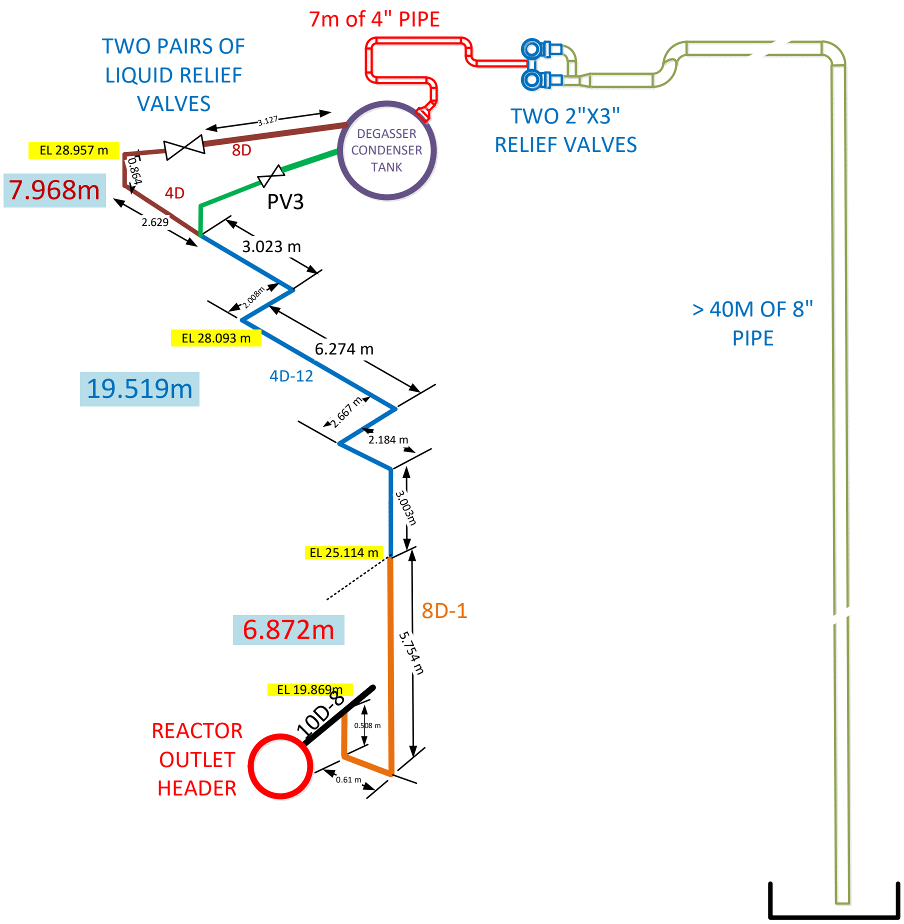


Figure 3: Overpressure protection system layout at a CANDU-6 plant

3. **Relief valves specifications require significant overpressure to discharge even the already low relief capacity of liquid or steam.**

Combined with the large pressure drop in the inlet line and the large 10% design accumulation (additional pressure rise to when valve disks fully open), the system cannot meet the ASME design requirements for an overpressure less than 10% for design basis accidents and 20% for lower frequency unanticipated accidents. Safety requirements dictate that for a PWR, the relief valve steam discharge capacity must be attained within a low-accumulation over set pressure (3% according to ASME BPV Code), and there be a generous liquid/steam relief capacity that envelopes relief requirements for both short- and long-term mismatches between heat generation and heat removal. For example, these mismatches can be from feedwater/steam line failures considered within the design basis as well as other long-term losses of boilers as heat sinks including those potentially leading to beyond design basis accidents. This always results in requirement for energy equivalent steam discharge for anything more than for an impulse, short term liquid only discharge, for overpressure protection.

4. **The two '50%' relief valves 40m away from a reactor header and subject to chattering and vibrations do not provide any redundancy and violate the requirements for safety system protection against single failures.**

Modern engineering practices for safety systems always include redundancy and periodic testing of passive relief valves for overpressure protection. Even with an inadequate, low specification of what constitutes a '100%' relief capacity requirement, installing two '50%' relief valves is questionable. An absence of a comprehensive and periodic inspection program like the one started by NRC for US LWRs (references v, vi, vii) could have avoided the 1994 event at Pickering where the relief valve was later examined and found to have been damaged.

5. **The relief valves are not isolated from the degasser condenser by a water seal, as is the practice at PWRs, and are subject to degradation and loss of function over time due to corrosive moisture laden environment.**

The relief valves are not isolated from degasser condenser gas/vapor environment that is likely very corrosive. They are also not subject to any periodic examination.

6. **A number of subsections of ASME section III, NB7000 are violated.**
   These relate to clear ASME Boiler & Pressure Vessel Code section 7000 guidelines for fluid relief capacity with consideration of heat loads and flashing, in-path obstructions, installation location and redundancy. See appendix B.

SAFETY SIGNIFICANCE

There are design basis accidents such as feedwater line breaks that have happened in other reactors that can lead to substantial over-pressurization on power. In a CANDU reactor feedwater loss has greater safety significance than in a PWR because each boiler serves a dedicated segment of core unlike PWRs where multiple secondary loops serve the same core and can dampen the effect. An asymmetrical feedwater break can lead to significant pressurization and early failures in at one of many components of the PHTs or in any of the interfacing systems or boiler tubes (Reference [xii]). We know that boiler tubes degrade in multiple ways and cannot be allowed to be over-pressurized beyond the thoughtful engineering analyses in OPR that recognizes boiler tube vulnerabilities. Boiler tube degradation data is abundant.

A station blackout at a CANDU station will likely cause a rupture in the pressure boundary with potential off-site releases of radioactivity. In addition, a longer than two-hour station blackout at Darlington and Bruce reactors can cause a relocation of primary inventory into the low-lying pressurizer and make ineffective all efforts to restore core cooling by addition of water inventory to boilers. The resulting overpressure, if not mitigated by an adequate steam relief through safety relief valves, properly redesigned and mounted, will likely cause an expensive loss of pressure boundary integrity and an un-necessary off-site dose with the reactor cooling system broken. An early pressure boundary failure due to an unmitigated overpressure will subdue the effectiveness of most measures the operator can take in managing a station blackout.

Except for an initial shot term discharge of any previously stagnant water in the leading pipes leading to the valves, the hot saturated or even subcooled pressurized HTS water will flash in the relief valves and thus valves should be designed and tested for at least two-phase water-steam relief and not just subcooled liquid water. Depending upon the piping downstream of the relief valve and the valve geometry that defines the back pressure, flashing for a saturated liquid D_2O at a 10% accumulation to 11 MPa may be as high at 47% with back pressure at atmospheric conditions.

The dominant driving forces in further lifting a spring-loaded relief valve are pressure, fluid drag and expansion. While a spring-loaded valve will begin to lift for any fluid at set pressure, less dense and compressible fluids like steam would be inept at lifting a spring-loaded disk valve as much as a subcooled liquid would. Tests showed that the valves which are designed to fully open for subcooled water did so and opened about 70% for air and between 1 to 20% for steam (References i, [xiii]). Steam relief capacity for valves designed and tested for liquid water must never be attained by any extrapolation or assumption of full valve lift. ASME BPV Section III requires that valves for steam service be tested in steam. Given the risk significance of safety relief valves, periodic testing of valve functions in steam is a must.

The utilities were required in the past by CNSC to justify the present design and what they presented appears to be <u>2 to 10 times more than what is theoretically possible</u> or tested steam relief capacity. For example as design steam relief capacity of both Pickering-B relief valves is 2*1.5=3 kg/s and the tested (Reference viii) relief capacity is 40+80=120 g/s, the industry claimed 30 kg/s (Reference [xiv]) which is significantly higher than even what was calculated with an assumed 100% lift as theoretical upper limits for choked sonic flow through the licensee claimed acceptable pressure rise to 12 MPa at BCRV which already is an unacceptable 40% pressure rise over the setpoint. Once the pressure drop in the lines leading to the relief valves is added, the industry claims of adequacy of design become un-supportable. For Bruce valves the design relief capacity is 1.5+1.5=3.0 kg/s while the tested relief capacity is between 0.5 and 2.0 kg/s (Reference xiii), but the claims of 13 kg/s at an elevated 12 MPa are twice what is theoretically possible even if the claimed full lift at 12 MPa is credited.

As a result of this current design of the overpressure protection system, even design basis accidents such as a loss of boilers as heat sinks can cause stresses that exceed service level C as in Pickering Safety Report [reference [xv]] where section 3.4.2.1 admits that "*The capacity of the bleed condenser relief valves is significantly lower than the HT LRVs, the HT system pressure increases rapidly. The peak ROH pressure is 16.3 MPa(a), which is below the applicable Level D service limit, as explained in Section 3.4.1*."

The design deficiency was long recognized within AECL and largely corrected in ACR-700 (reference xvi) where the steam relief capacity of the DCRVs was 10 times larger at 21.3 kg/s per valve compared to most operating CANDUs and was the primary design specification and not an afterthought to a liquid relief capacity specification for design & ASME certification.

CONCLUSIONS & RECOMMENDATION

The overpressure protection in all current CANDU PHWR primary heat transport systems needs to be strengthened. We recommend that a consultative process be undertaken with all licensees and that we present our findings to them.

APPENDIX B – Schematics of typical PWR and CANDU PHTS pressure relief designs

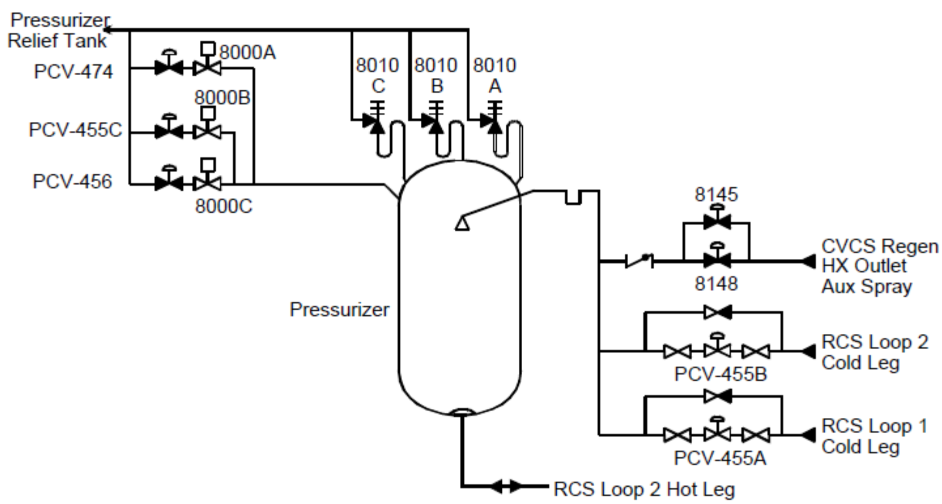


Figure 4 : TYPICAL PWR PRESSURE RELIEF VALVE INSTALLATION - DITECT AND CLOSE TO PRESSURIZER

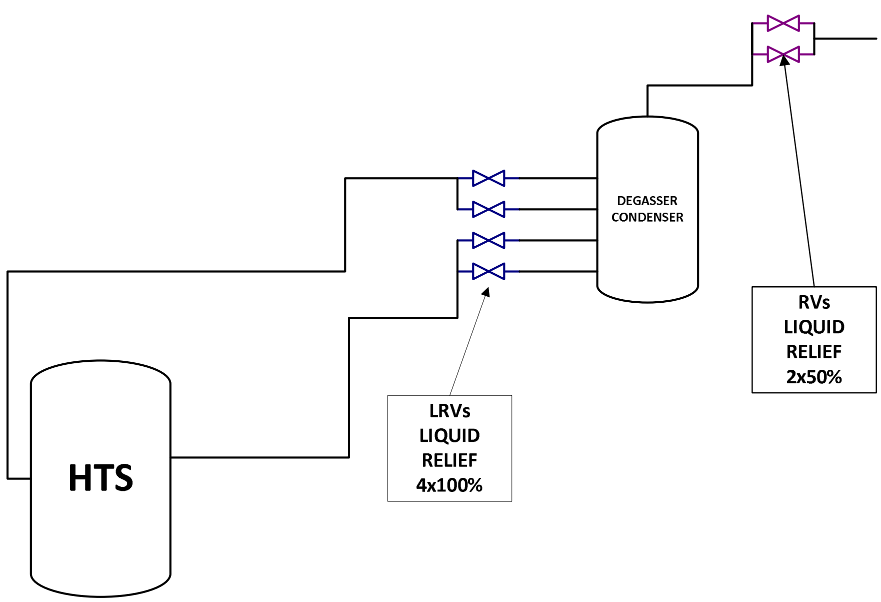
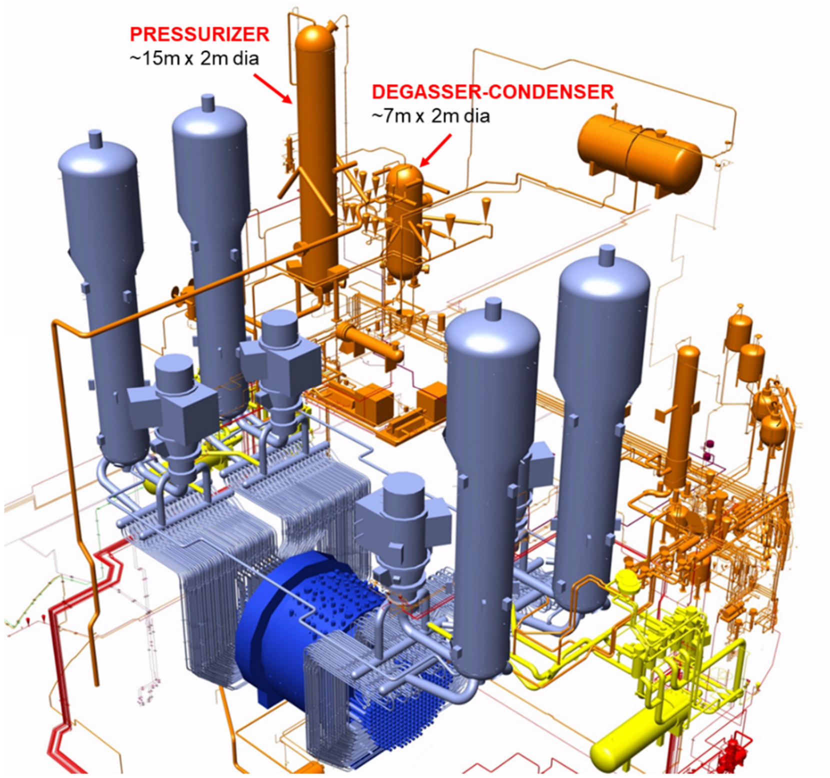
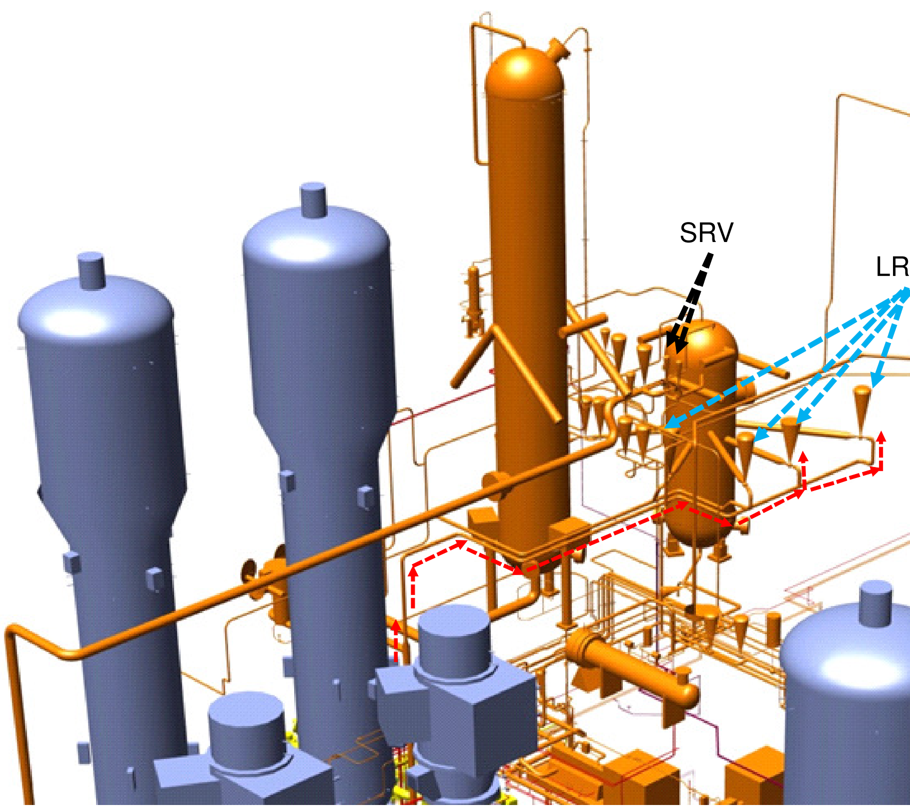


Figure 5: TYPICAL PHWR CANDU PHTS PRESSURE RELIEF – INDIRECT AND AWAY FROM PHTS ; DISCHARGE PATH INTERRUPTED BY OTHER VALVES, VESSELS & SYSTEMS

Appendix B : Installation geometry of relief valves for CANDU 6 reactors (Darlington relief valve layout different due to lower placement of degasser condenser)





Appendix C : Notes on Specifics of design that are inconsistent with ASME BPV Code

	CANDU DESIGN AND FINDINGS
NB-7140 INSTALLATION; *NB-7141 Pressure Relief Devices* (a) Pressure relief devices shall be as close as practicable to the major source of overpressure.	Relief valves are 40m away leading to substantial pressure rise upon valve actuation and substantial valve chatter in steam discharge as noted in Pickering; typical installation in PWRs is within a few meters of pressurizers.
(b) The connection between a system and its pressure relief device shall have a minimum inside diameter equal to or greater than the nominal inside diameter of the pressure relief device inlet. <u>The opening in the connection shall be designed to provide direct and unobstructed flow between the system and the pressure relief device.</u>	Relief valves are obstructed by LRVs and DCRVs and are not unobstructed. LRVs are not classic relief valves that would open on direct action of pressure. These are pilot operated spring-loaded valves kept closed by pneumatic pressure and opened by a signal. Their erroneous opening has caused a number of LOCAs. Their remaining shut-in absence of electric power when required in a SBO can be catastrophic.
(d) The connection between a system and its safety relief valve or relief valve shall not result in accumulated line losses greater than 3% of the relieving pressure.	Line losses are > 3% and are between 10% and 20% for design liquid discharge and industry claimed steam discharge.
(f) … Back pressure that may exist or develop shall not reduce the relieving capacity of the relieving device(s) below that required to protect the system; potential for flashing shall be considered.	Substantial back pressure likely caused by long pipes downstream of the relief valves Very important to note that flashing of hot, saturated primary fluid not considered in relief valves erroneously sized for liquid discharge
(g) Valve installation not in accordance with (c), (d), (e), and (f) above may be used provided: (1) the NV Certificate Holder confirms that the valve design is satisfactory for the intended installation and satisfies the requirements of the valve Design Specification; (2) the valves are adjusted for acceptable performance in conformance with the requirements of the valve Design Specification; (3) technical justification for the adequacy of the installation is provided in the Overpressure Protection Re-port, including verification that the requirements of (1) and (2) have been met.	No justification provided or possible in light of the safety requirements for limiting overpressure to less that 10% for design basis accidents or 20% for unanticipated events.
NB-7310 EXPECTED SYSTEM PRESSURE TRANSIENT CONDITIONS; *NB-7311 Relieving Capacity of Pressure Relief Devices*	The design documents do not acknowledge the distant placement of valves or consider the substantial system pressure rise that will be caused by fluid pressure losses due to flow through over 40m of pipe with a dozen bends, other valves (LRV)

(a) The total relieving capacity of the pressure relief devices … shall take into account any losses due to flow through piping and other components.	and vessels (DC, PZR) on the way from header to the relief valves and ~30m of pipe from relief valves to sump.
(b) The total relieving capacity shall be sufficient to prevent a rise in pressure of more than 10% above the Design Pressure of any component within the pressure-retaining boundary of the protected system under any expected system pressure transient conditions as summarized in the Overpressure Protection Report	Relief capacity for liquid too small to compensate for liquid swell at any time before 1000 seconds after reactor trip or vapour generation at any time at all. Examples abundant in safety reports (e.g., on a loss of feedwater from 1 steam generator and potential pressure rise as high as 100%) and the by steam production at decay powers on a loss of heat sinks.
NB-7320 UNEXPECTED SYSTEM EXCESS PRESSURE TRANSIENT CONDITIONS; *NB-7321 Relieving Capacity of Pressure Relief Devices* For the cases where the pressure transients fall under the guise of 'unexpected system excess pressure transient conditions' subsection NB-7321 further adds the requirement that "the system overpressure established for setting the required total relieving capacity … shall be such that the calculated stress intensity and other design limitations for Service Limit C specified in NB-3000 are not exceeded for each of the components in the protected system	Steam relief capacity too small to limit uncontrolled pressure rise in the HTS on a loss of heat sinks and a loss of primary circulation in a station blackout scenario. It is easy to envision stagnated fuel channel creating a liquid swell at the rate of steam production consistent with channel power after about 1 hour. Hence a need to directly relieve that volumetric amount of steam production/liquid swell on a loss of heat sinks.
NB-7200 OVERPRESSURE PROTECTION REPORT *NB-7220 CONTENT OF REPORT* The Overpressure Protection Report shall define the protected systems and the integrated overpressure protection provided. As a minimum, the Report shall include the following:	
(a) drawings showing arrangement of protected systems, including the pressure relief devices; (b) the range of operating conditions, including the effect of discharge piping back pressure;	Not made available. Not reported to have been considered
(a) the redundancy and independence of the pressure relief devices and their associated pressure sensors and controls employed to preclude a loss of overpressure protection in the event of a failure of any pressure relief device, sensing elements, associated controls, or external power sources;	There is no redundancy on RVs – RV11, RV 21 on PLGA – two 50% valves, 3with definition of 100% in error No consideration of loss of electronic information from pressure sensors, especially after a SBO.

NB-7240 REVIEW OF REPORT AFTER INSTALLATION (a) Any modification of the installation from that used for the preparation of the Overpressure Protection Report shall be reconciled with the Overpressure Protection Report. (b) Modifications shall be documented in an addendum to the Overpressure Protection Report. The addendum shall contain a copy of the as-built drawing and shall include either: (1) a statement that the as-built system meets the requirements of the Overpressure Protection Report; or (2) a revision to the Overpressure Protection Report to make it agree with the as-built system; or (3) a description of the changes made to the as-built system to make it comply with the Overpressure Protection Report.	Not information on 'as built' system in any industry document. If the as built configuration was considered, the pressure drop in the long piping would have been an obvious hindrance to its acceptability.

[i] 98-01347-OPR-000 - Overpressure Protection Report - Qinshan CANDU Project

[ii] NUREG/CR-7110-vol2 - State-of-the-Art Reactor Consequence Analyses Project Volume 2: Surry Integrated Analysis – section 4.2.1 referring to Surry Updated Final Safety Analysis Report, Revision 38, 05/31/07.

[iii] PWR Safety and Relief Valve Adequacy Report For Georgia Power Company Alvin W. Vogtle Unit 1 and Unit 2, May 1985

[iv] EGG-RTAP-I0627, Technical Evaluation Report, TMI Action--NUREG-0737(II.D.I), Relief And Safety Valve Testing Comanche Peak - Unit 2, DOCKET NO. 50-446

[v] NUREG-0578 - Section 2.1.2 -TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations

[vi] NUREG-0737 - Item II.D.1.A - Clarification of TMI Action Plan Requirements

[vii] EPRI Valve Test Program Staff, EPRI PWR Safety and Relief Test Program, Safety and Relief Valve Test Report, EPRI NP-2628-SR, December 1982.

[viii] Pickering NGS 'B' Replacement Bleed Condenser Relief Valves (33320RV 108 & RV163) ,Testing and Design Summary Report , NK30-DRT-33323-00005

ix PICKERING REPORT NK30-DRT-33323~10002REVOOO . UNIT 058 , System: Valves , Title: Design Report Test Report, Water & Steam Flow on 2" X3 " and 4" X 6" B & R Water Relief Valves. Condensed Report for Ontario Power Generation, 2001

x Bleed Condenser Relief Valves NK30-TS-33320-0001

xi Design Requirements Replacement Bleed Condenser Reilef Valves NK30-DR- 33320-00001

[xii] Pickering A safety report

xiii AECL TTR-638 (Bruce Relief valve test report) xiv File: N-00531-P CD# N-CORR-00531-02663, P. R. Charlebois to Mr. Schaubel and Ms. Ecroyd, Sustained loss of all heat sinks, 28 August 2003

xv Pickering Nuclear 1-4 Safety Report: Part 3 - Accident Analysis NA44-SR-01320-00002-R004; 2013-10-31

[xvi] Preliminary Overpressure Protection Assessment of Major Systems System, ACR-700 , 10810-01347-ASD-001 Revision 0 (section 9.5.1.1)